DCPP L181 NRC Exam **6 March 2020**

Multiple Choice (Fill In Your Choice) NAME: _**ANSWER KEY**_____

If you change your answer, write your selection in the blank and initial.

No.					
1.	A	B	C	●	____
2.	●	B	C	D	____
3.	●	B	C	D	____
4.	A	B	C	●	____
5.	A	B	●	D	____
6.	A	B	●	D	____
7.	A	●	C	D	____
8.	●	B	C	D	____
9.	A	●	C	D	____
10.	A	●	C	D	____
11.	A	B	●	D	____
12.	A	B	●	D	____
13.	●	B	C	D	____
14.	A	B	C	●	____
15.	A	●	C	D	____
16.	A	B	C	●	____
17.	A	●	C	D	____
18.	A	●	C	D	____
19.	A	●	C	D	____
20.	A	B	C	●	____
21.	A	●	C	D	____
22.	●	B	C	D	____
23.	●	B	C	D	____
24.	A	B	●	D	____
25.	A	●	C	D	____
26.	●	B	C	D	____
27.	A	●	C	D	____
28.	●	B	C	D	____
29.	A	●	C	D	____
30.	●	B	C	D	____
31.	A	●	C	D	____
32.	●	B	C	D	____
33.	A	B	●	D	____
34.	A	●	C	D	____
35.	●	B	C	D	____
36.	A	B	C	●	____
37.	A	B	●	D	____
38.	A	B	C	●	____
39.	●	B	C	D	____
40.	A	B	●	D	____
41.	A	B	C	●	____
42.	A	●	C	D	____
43.	A	B	C	●	____
44.	A	B	C	●	____
45.	A	B	C	●	____
46.	A	B	C	●	____
47.	●	B	C	D	____
48.	A	●	C	D	____
49.	A	B	C	●	____
50.	A	●	C	D	____

Multiple Choice (Fill In Your Choice) NAME: **ANSWER KEY**

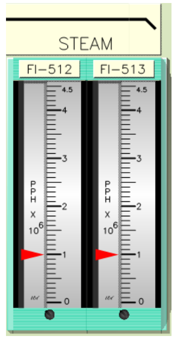
If you change your answer, write your selection in the blank and initial.

51.	A	B	●	D	____	**76.**	A	B	●	D	____
52.	A	B	C	●	____	**77.**	A	●	C	D	____
53.	●	B	C	D	____	**78.**	●	B	C	D	____
54.	A	●	C	D	____	**79.**	●	B	C	D	____
55.	●	B	C	D	____	**80.**	A	B	C	●	____
56.	A	B	●	D	____	**81.**	A	B	●	D	____
57.	A	B	●	D	____	**82.**	A	B	C	●	____
58.	A	●	C	D	____	**83.**	A	●	C	D	____
59.	A	B	●	D	____	**84.**	A	●	C	D	____
60.	A	B	C	●	____	**85.**	A	B	●	D	____
61.	A	B	C	●	____	**86.**	A	B	C	●	____
62.	A	B	●	D	____	**87.**	A	●	C	D	____
63.	A	●	C	D	____	**88.**	●	B	C	D	____
64.	A	B	●	D	____	**89.**	●	B	C	D	____
65.	A	●	C	D	____	**90.**	A	B	C	●	____
66.	A	●	C	D	____	**91.**	A	●	C	D	____
67.	A	B	●	D	____	**92.**	A	B	●	D	____
68.	●	B	C	D	____	**93.**	A	●	C	D	____
69.	A	●	C	D	____	**94.**	●	B	C	D	____
70.	●	B	C	D	____	**95.**	●	B	C	D	____
71.	A	B	C	●	____	**96.**	A	B	●	D	____
72.	A	●	C	D	____	**97.**	A	B	●	D	____
73.	A	B	●	D	____	**98.**	A	B	C	●	____
74.	●	B	C	D	____	**99.**	A	B	C	●	____
75.	A	●	C	D	____	**100.**	A	●	C	D	____

Examination Outline Cross-Reference		**Level**	RO
003 K3.02 - Knowledge of the effect that a loss or malfunction of the RCPS will have on the following: S/G		**Tier #**	2
		Group #	1
		K/A #	003 K3.02
		Rating	3.5

Question 01

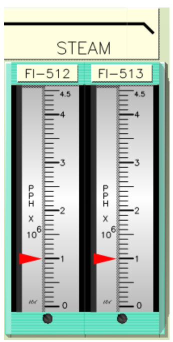
Unit 1 is at 30% power, with the following 1-1 Steam Generator steam flow indication:

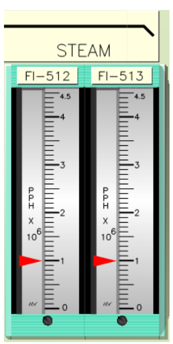


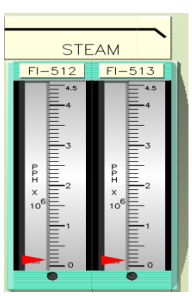
1-1 RCP trips. The plant remains at power.

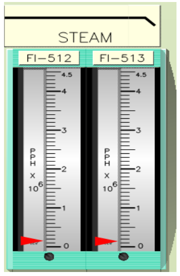
1) What will be the 1-1 Steam Generator steam flow indication shortly, (approximately 30 seconds), after 1-1 RCP trips?

2) Total steam flow will be ________________________ its initial value.

A. 1)  2) lower by approximately 25% from

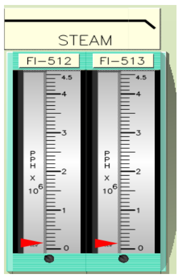
B. 1)  2) approximately the same as

C. 1)  2) lower by approximately 25% from

D. 1) 

2) approximately the same as

Proposed Answer:

D. 1) 

2) approximately the same as

Explanation:

A. Incorrect. Steam flow lowers, does not remain unchanged. Second part plausible as steam flow rises in the other 3 loops. Steam flow in the other 3 loops will rise (while it lowers to almost zero in the affected loop) and steam flow will return to its original value. Plausible to think that steam flow will be less overall.
B. Incorrect. Steam flow lowers. Second part correct.
C. Incorrect. First part correct, steam flow lowers. Second part is incorrect.
D. Correct. RCP trip will cause the loop temperature to lower and the steaming from the loop will stop (rises in the other 3 loops). The loss of steam flow from the loop will be picked up by the other loops.

Technical References: LTH-18

References to be provided to applicants during exam: None

Learning Objective: 10583 - DESCRIBE the reactor, RCS and Secondary System responses to each of the following transients: e. Stopping a Reactor Coolant Pump (RCP) with no resultant reactor trip.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.14	

Difficulty: 3.1

Examination Outline Cross-Reference		**Level**	RO
078 K4.03 - Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Securing of SAS upon loss of cooling water (note: apparent typo - written to address securing IAS as the KA is instrument air and SAS is its own KA- 079)		**Tier #**	2
		Group #	1
		K/A #	078 K4.03
		Rating	3.1

Question 02

A loss of Service Cooling Water (SCW) has occurred.

What Instrument Air compressor(s) cooling is/are affected by the loss of SCW?

A. 0-5 and 0-6

B. 0-6 and 0-7

C. 0-5 only

D. 0-7 only

Proposed Answer: A. 0-5 and 0-6

Explanation:

A. Correct. SCW supplies cooling to rotary AC 0-5 and 0-6 only, 0-7 is air cooled.
B. Incorrect. SCW cools rotary air compressors 0-5 and 0-6 (along with all 4 reciprocating air compressors). Plausible to believe 0-5 is the air cooled compressor.
C. Incorrect. SCW cools rotary air compressors 0-5 and 0-6 (along with all 4 reciprocating air compressors). Plausible to know that there is one AC that is different than the other two and think that the difference is only one is cooled by SCW (as opposed to only one not cooled by SCW) and that one is 0-5.
D. Incorrect. SCW cools rotary air compressors 0-5 and 0-6 (along with all 4 reciprocating air compressors). Plausible to know that -07 is different than the other two (air cooled) and think that the difference is that it is cooled by SCW (as opposed to only one not cooled by SCW).

Technical References: OVID 106725 sheet 2, LK-1

References to be provided to applicants during exam: None

Learning Objective: Describe Compressed Air System components. (7199)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.4	

Difficulty: 2.0

Examination Outline Cross-Reference	**Level**	RO
026 K4.04 - Knowledge of CSS *design feature(s)* and/or interlock(s) which provide for the following: Reduction of temperature and pressure in containment after a LOCA by condensing steam, to reduce radiological hazard, and protect equipment from corrosion damage (spray)	**Tier #**	2
	Group #	1
	K/A #	026 K4.04
	Rating	3.7

Question 03

Concerning the design of the Containment Spray system:

1) Proper pH in the Recirc Containment Sump following a LOCA is assured by:
2) Design Containment pressure will not be exceeded if at least one Containment Spray train and a minimum of _____ CFCUs operate to remove heat and condense steam following a LOCA.

A. 1) an initial minimum Containment Spray Additive Tank level of 60%.
   2) 2

B. 1) an initial minimum Containment Spray Additive Tank level of 60%.
   2) 3

C. 1) the Containment Spray pumps remaining aligned to the RWST until level is 33%.
   2) 2

D. 1) the Containment Spray pumps remaining aligned to the RWST until level is 33%.
   2) 3

Proposed Answer: A. 1) an initial minimum Containment Spray Additive Tank level of 60%. 2) 2

Explanation:

A. Correct. The purpose of the SAT is to deliver with one Containment Spray pump running, enough sodium hydroxide (NaOH) into Containment to achieve a minimum pH of 8.0 in the recirculation sump prior to reaching the RWST low-low level. The basic pH helps the spray droplets entrain gaseous fission products, particularly iodine, aids in keeping fission products in solution in the recirculation sump and helps mitigate chloride stress corrosion of austenitic stainless-steel materials inside Containment.
   One train of Containment Spray and two of five CFCUs provide sufficient heat removal to maintain Containment pressure below its design value of 47 psig following a design basis LOCA or MSLB.
B. Incorrect. First part is correct. Second part is incorrect, 3 CFCUs is the minimum number of CFCUs in Technical Specifications without requiring LCO action.
C. Incorrect. Second part is correct. First part is incorrect. While the pumps remain aligned to the RWST until 4%. 33% is the level at which cold leg recirc is aligned (RHR pumps trip at 33%).
D. Incorrect. Both parts incorrect. First part is incorrect. The pumps remain aligned to the RWST until 4%. Second part incorrect, 3 CFCUs is the minimum number of CFCUs in Technical Specifications without requiring LCO action.

Technical References: LI-6, LCO 3.6.6

References to be provided to applicants during exam: None

Learning Objective: Explain significant Containment Spray System design features and the importance to nuclear safety. (40802)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.8	

Difficulty: 2.6

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
059 A2.07 - Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Tripping of MFW pump turbine	**Group #**	1
	K/A #	059 A2.07
	Rating	3.0

Question 04

Unit 1 is at 100% power.

Main Feedwater pump 1-1 trips. An automatic ramp does not occur.

In accordance with OP AP-15, Loss of Feedwater Flow,

1) an automatic ramp of _______________ should be occurring.

2) the operator will ____.

A. 1) 40 MW/Min
   2) manually initiate the ramp

B. 1) 40 MW/Min
   2) trip the reactor

C. 1) 225 MW/Min
   2) manually initiate the ramp

D. 1) 225 MW/Min
   2) trip the reactor

Proposed Answer: D. 1) 225 MW/Min 2) trip the reactor

Explanation:

A. Incorrect. 40 MW/Min is the ramp rate for the loss of the heater drip pump. The action is to trip the reactor if the ramp does not initiate automatically.
B. Incorrect. Ramp rate is incorrect. Action is correct.
C. Incorrect. Ramp rate is correct. Action is to trip the reactor if the ramp does not occur.
D. Correct. Ramp rate for trip of a feed pump is 225 MW/Min. If the ramp does not occur, the operator trips the reactor per the immediate actions of OP AP-15.

Technical References: OP AP-15, section A, OP1.DC10

References to be provided to applicants during exam: None

Learning Objective: State the steps and transitions in procedures that are considered immediate actions. (9693)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Difficulty: 2.4

Examination Outline Cross-Reference		**Level**	RO
		Tier #	2
010 A3.02 - Ability to monitor automatic operation of the PZR PCS, including: PZR pressure		**Group #**	1
		K/A #	010 A3.02
		Rating	3.6

Question 05

GIVEN:
- A rapid downpower from 100% to 75% has just occurred on Unit 1
- Pressurizer pressure channels indicate as follows:
  - PI – 455 – 2220 psig
  - PI – 456 – 2270 psig
  - PI – 457 – 2235 psig
  - PI – 474 – 2210 psig

What is the status of the Pressurizer Heaters controlled by HC-455K, Pzr Press Control?

A. All the heaters are off

B. Only the Backup heaters are on

C. Only the Proportional heaters are on

D. Both the Proportional and Backup heaters are on

Proposed Answer: C. Only the Proportional heaters are on

Explanation:
A. Incorrect. The pressure channel in control is the second highest. This is PI-457, at 2235 psig. This is high enough that proportional heaters would be on. Plausible because if its thought the highest channel is in control, this would be the answer.
B. Incorrect. Plausible if its thought the backup heaters that are on and not the proportional heaters.
C. Correct. At 2235 psig, the proportional heaters are on and backup heaters are off.
D. Incorrect. This would be correct if PI-474, the lowest channel, was in control.

Technical References: OIM page A-4-6
References to be provided to applicants during exam: None
Learning Objective: 4560 - Describe the operation of the Pzr, Pzr Pressure and Level Control System

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank # 31 DCPP L111 11/2012	X
	New	
	Past NRC Exam DCPP NRC 11/2012	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.9

Examination Outline Cross-Reference	Level	RO
	Tier #	2
061 K6.02 - Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps	Group #	1
	K/A #	061 K6.02
	Rating	2.6

Question 06

Unit 2 tripped from 100% power.

2-3 Motor Driven AFW Pump starts and immediately trips.

Steam Generator AFW Supply Valves, LCV-115 and LCV-113 will:

A. Close due to the pump trip.

B. Close due to low pump discharge pressure.

C. Open due to the pump trip.

D. Open due to low pump discharge pressure.

Proposed Answer: C. Open due to the pump trip.

Explanation:

A. Incorrect. The valves will fail open. Plausible to think the valves would close if the pump is not running.
B. Incorrect. The valves will fail open. Plausible, this is the response of the valves on low pump discharge pressure if the pump is running (to prevent pump runout).
C. Correct. Opening the pump breaker removes power from the valves and they go full open. This is the normal state of the valves when the pumps are not running.
D. Incorrect. The valves fail open. Plausible that the valves would open in an attempt to raise system pressure.

Technical References: LD-1

References to be provided to applicants during exam: None

Learning Objective: 37635- Describe controls, indications, and alarms associated with the Auxiliary Feedwater System.

Question Source:	Bank #18 DCPP NRC 1/2010	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam: DCPP 1/2010	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.2

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
008 A3.02 - Ability to monitor automatic operation of the CCWS, including: Operation of the CCW pumps, including interlocks and ~~the CCW booster pump~~(N/A)	**Group #**	1
	K/A #	008 A3.02
	Rating	3.2

Question 07

The Standby Select switch for a non-running CCW pump is in MANUAL.

What automatic start(s) is/are still available?

1) Safety Injection
2) Transfer to Diesel
3) Low System Pressure

A. 1 only

B. 1 and 2

C. 3 only

D. 2 and 3

Proposed Answer: B. 1 and 2

Explanation:
All the listed possibilities are trips of the CCW pump and therefore plausible if its not known which require the MANUAL/AUTO switch to be in AUTO to be an active trip.

A. Incorrect. Auto is not required for the pump to start in response to a SI, however, a pump will also start on a transfer to diesel.
B. Correct because Auto is not required for the pump to start in response to a SI or Transfer to Diesel.
C. Incorrect. The Standby Select switch If in Auto, the pump will start in response to low system pressure, but will not if in Manual. Some equipment does not start if not in Auto, such as a diesel.
D. Incorrect because Auto is not required for the pump to start in response to a Transfer to Diesel but is for low system pressure.

Technical References: LF-2

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the CCW System. (35487)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #7 DCPP NRC Exam 10/2016	X
	New	
	Past NRC Exam	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.0

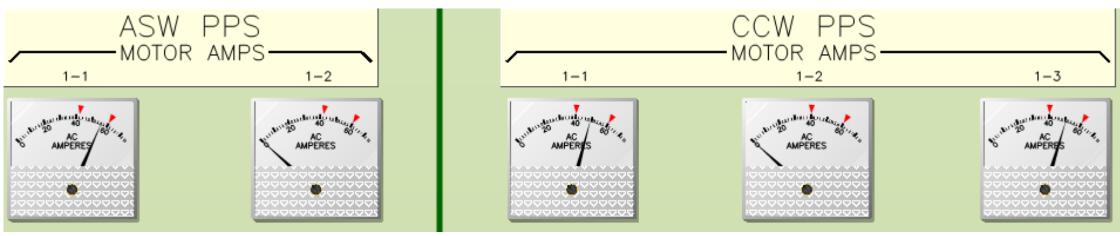
Examination Outline Cross-Reference		**Level**	RO
		Tier #	2
076 A4.04 Service Water (Aux Saltwater – DCPP equivalent)		**Group #**	1
Ability to manually operate and/or monitor in the control room:		**K/A #**	076 A4.04
Emergency heat loads		**Rating**	3.5

Question 08

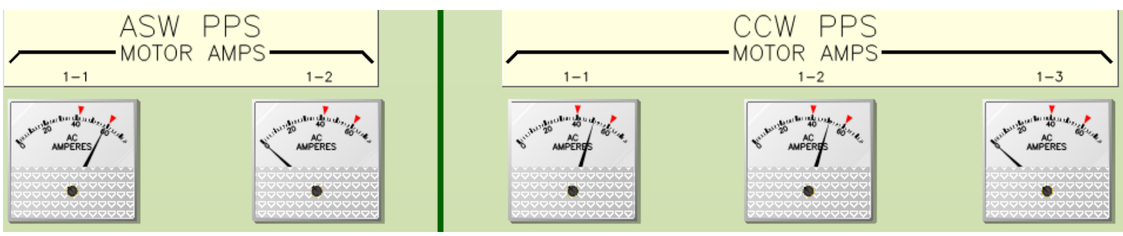
GIVEN:
- Safety Injection actuates on Unit 1 from 100% power
- Following the reactor trip, 4 kV Bus G de-energizes
- The crew is performing EOP E-0, Reactor Trip or Safety Injection

When the operator is performing Appendix E, ESF Auto Actions, Secondary and Auxiliaries Status, what will be the ASW and CCW pump status?

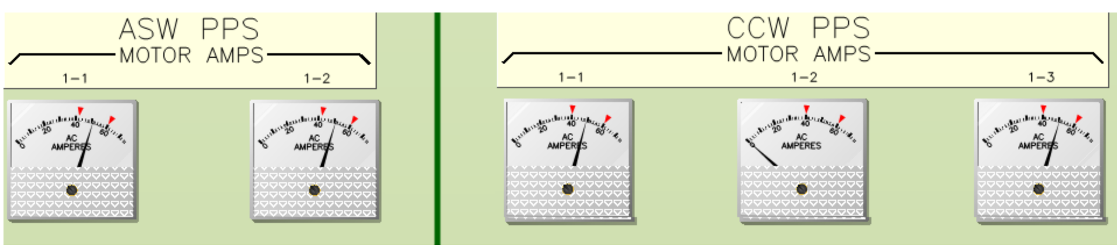
A.



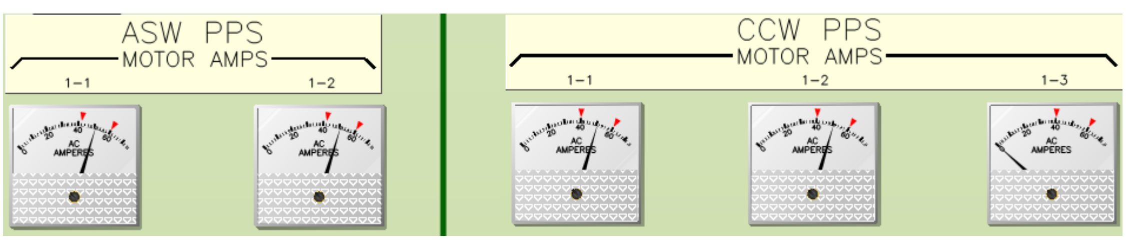
B.



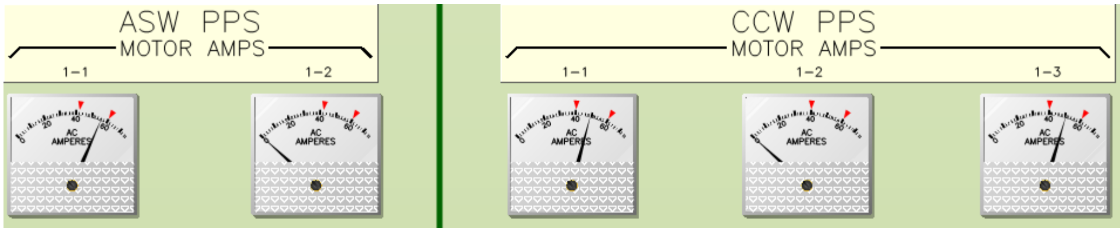
C.



D.



Proposed Answer:

A.	

Explanation:
Note, addressing the "monitoring" of ASW and the emergency loads for ASW – the CCW system. Operator must know what pumps should be running and which are affected by the loss of the vital bus.

A. Correct. Normally, all ASW and CCW pumps start on SI. Bus G powers ASW pump 1-2 and CCW pump 1-2 and therefore, will not be running.
B. Incorrect. First part is correct. Second part is plausible – the power supplies do not align that pumps 1, 2 and 3 are from bus F, G and H. For instance, CCP 1-3 is powered from bus G and RHR pumps 1-1 and 1-2 are powered from G and H.
C. Incorrect. First part is plausible as there are instances of systems with 2 pumps that do not have a pump on bus G, such as SI. Second part is correct.
D. Incorrect. First part is plausible as there are instances of systems with 2 pumps that do not have a pump on bus G, such as RHR. Second part is plausible – the power supplies do not align that pumps 1, 2 and 3 are from bus F, G and H. For instance, CCP 1-3 is powered from bus G and RHR pumps 1-1 and 1-2 are powered from G and H

Technical References: OIM J-1-1 and J-6-1
References to be provided to applicants during exam: None
Learning Objective: State the power supplies to CCW System components. (8129)
State the power supplies to ASW System components. (5339)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.2

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
006 G2.2.39 ECCS: Knowledge of less than or equal to one hour Technical Specification action statements for systems.	**Group #**	1
	K/A #	006 G2.2.39
	Rating	3.9

Question 09

Unit 1 is at 100% power.

Which of the following would require the crew to enter an LCO with a COMPLETION TIME of one hour or less?

1. Nitrogen pressure in two Accumulators is less than the pressure required by LCO 3.5.1, Accumulators
2 RWST level is less than the level required by LCO 3.5.4, Refueling Water Storage Tank, (RWST)
3. Reactor coolant pump seal injection flow resistance is less than the 0.2117 ft/gpm^2 required by LCO 3.5.5, Seal Injection Flow

A. 1 only

B. 1 and 2

C. 3 only

D. 2 and 3

Proposed Answer: B. 1 and 2

Explanation:

A. Incorrect. Only partially correct. Two inoperable accumulators require action within one hour (immediately enter LCO 3.0.3) However, Low RWST level requires action to restore the RWST to OPERABLE status within one hour.
B. Correct. Two accumulators inoperable, regardless of the reason, is an immediate (enter LCO 3.0.3) action. Additionally, low level in the RWST is a one hour action, (restore to OPERABLE) action
C. Incorrect. Seal flow resistance is 4 hours. Plausible because a reduction of available ECCS injection could be viewed as needing immediate action.
D. Incorrect. First part is correct.

Technical References: LCO 3.5.1, 3.5.4, 3.5.5

References to be provided to applicants during exam: None

Learning Objective: 9697E - Apply TS 3.5 Technical Specification LCOs

Question Source:	Bank # DCPP	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Difficulty: 3.1

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
063 K1.02 - Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems: AC electrical system	**Group #**	1
	K/A #	063 K1.02
	Rating	2.7

Question 10

Unit 1 is at 100% power.

What is the effect of opening the DC supply breaker to a vital 120 VAC Inverter?

1. PK19-18, VITAL UPS TROUBLE alarm
2. PK19-19, VITAL UPS FAILURE alarm
3. Lowering of Inverter output voltage

A. 1 only

B. 2 only

C. 1 and 3

D. 2 and 3

Proposed Answer: B. 2 only

Explanation:
Question tests the physical connection between the DC system and the (120 VAC) electrical distribution system. The battery (DC) is the backup to the inverter. The AC input is at a slightly higher voltage. If the DC is lost, the higher AC is still supplying the inverter, so no change involtage will occur. However the loss of DC does make the inverter inoperable and therefore, is a Failure, not Trouble alarm.

A. Incorrect. The Failure, not Trouble alarm is generated.
B. Correct. The higher AC voltage is unaffected. The loss of DC results in a Failure alarm only.
C. Incorrect. Both parts are incorrect.
D. Incorrect. First part is correct. Second part incorrect..

Technical References LJ-10
References to be provided to applicants during exam: None
Learning Objective: State the power supplies to Instrument AC System components. (3345)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Difficulty: 3.0

Examination Outline Cross-Reference		**Level**	RO
		Tier #	2
026 K2.01Knowledge of bus power supplies to the following: Containment spray pumps		**Group #**	1
		K/A #	026 K2.01
		Rating	3.4

Question 11

GIVEN:

- A loss of offsite power and LOCA have occurred on Unit 1
- Containment Spray has actuated

Which Unit 1 Emergency Diesel Generators (EDG) will be powering the Containment Spray pumps?

1) Containment Spray pump 1-1 will be powered from EDG _____.

2) Containment Spray pump 1-2 will be powered from EDG _____.

A. 1) 1-1
   2) 1-2

B. 1) 1-1
   2) 1-3

C. 1) 1-2
   2) 1-1

D. 1) 1-2
   2) 1-3

Proposed Answer: C. 1) 1-2 2) 1-1

Explanation: (unit difference)
Unit 1 EDGs 1-1 powers bus H/ EDG 1-2 powers bus G/ EDG1-3 powers bus F
Unit 2 EDGs 2-1 powers bus G/ EDG 2-2 powers bus H/ EDG 2-3 power bus F
Containment spray pumps are powered from buses G and H (both units)

A. Incorrect. This would be for Unit 2 -EDGs 2-1 (Bus G) and 2-2 (Bus H) power the spray pumps.
B. Incorrect. Plausible EDG is Bus F and powers 1-1. 1-3 power bus F but there is no spray pump on F.
C. Correct. For Unit 1, the pumps are powered from Bus G (EDG 1-2) and H (1-1)
D. Incorrect. First part correct. Bus F (EDG 2-3) does not power a spray pump. Plausbile that 1-3 powers bus H, if its thought the buses F, G, H are powered from EDG 1-1, 1-2 and 1-3 respectively.

Technical References: OIM J-1-1
References to be provided to applicants during exam: None
Learning Objective: 6022 - State the power supplies to CSS components

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #40 DCPP NRC 07/2011	X
	New	
	Past NRC Exam DCPP 07/2011	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	

	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.8	
Difficulty:2.1		

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
039 K3.03 - Knowledge of the effect that a loss or malfunction of the MRSS will have on the following: AFW pumps	**Group #**	1
	K/A #	039 K3.03
	Rating	3.2

Question 12

Unit 1 is at 10% power.

A steam break, inside containment, occurs on Steam Generator 1-2. Safety Injection actuates. Narrow range level on Steam Generator 1-2 is off scale low. Other steam generator narrow range levels are 65%.

1) AFW pump 1-1__________________ automatically start.

2) After the steam supply to AFW pump 1-1 is isolated in accordance with EOP E-2, Faulted Steam Generator Isolation, AFW pump 1-1 is ______________ of providing 100% of its rated AFW flow.

A. 1) will
   2) capable

B. 1) will
   2) NOT capable

C. 1) will NOT
   2) capable

D. 1) will NOT
   2) NOT capable

Proposed Answer: C. 1) will NOT 2) capable

Explanation:

A. Incorrect. Plausible the TDAFW pump would start but unlike the Motor Driven AFW pumps, is not started by SI. It is started by 2 of 4 steam generators (the motor driven AFW pumps start on 1 of 4), AMSAC or 12 kV UV (all MDAFW pumps starts). Second part is correct. While the two leads feed a common line to the suction of the TDAFW pump, the steam supplies are 100% redundant and available flow from the AFW pump is unaffected by the closing of one of the steam supplies.
B. Incorrect. The TDAFW pump is not started by SI. It is started by 2 of 4 steam generators (only one is low), AMSAC or 12 kV UV. Second part incorrect. While the leads from the steam generators feed a common header, the amount of flow from the TDAFW pump is unaffected by closing one, each one can supply 100% of the steam flow required to have the pump supply its rated flow.
C. Correct. SI starts the motor driven pumps start. The TDAFW pump will not automatically start. The steam supplies are redundant, only one is necessary for the pump to supply rated flow.
D. Incorrect. First part is correct. Second part is incorrect. The flow available is unaffected by the closing of one of the two steam supplies.

Technical References: OIM D-1-2, OVID 106704 sheet 4

References to be provided to applicants during exam: None

Learning Objective: State the purpose of Auxiliary Feed Water System components.

☐Turbine Steam Supply Control Valves FCV-37 and FCV-38

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.4	

Difficulty: 3.0

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
103 K4.06 Knowledge of containment system design feature(s) and/or interlock(s) which provide for the following: Containment isolation system	**Group #**	1
	K/A #	103 K4.06
	Rating	3.1

Question 13

What will directly cause Containment Ventilation Isolation (CVI) to automatically actuate?

1. SI signal present
2. Phase A due to high containment pressure
3. Phase B due to high containment pressure

A. 1 only

B. 2 only

C. 3 only

D. 1 and 3

Proposed Answer: A. 1 only

Explanation:

A. Correct. SI automatically actuates CVI. Manual actuation (not Containment pressure) of Phase A or Phase B will also cause CVI.
B. Incorrect. 3 psig isolates containment penetrations by initiating Phase A, but CVI by Phase A is by MANUAL actuation (1 of 2 pushbuttons) of Phase A.
C. Incorrect. 22 psig actuates Phase B, however, to actuate CVI on Phase B, manual Phase B actuation (2 of 2) is required.
D. Incorrect. This is the coincidence for actuating Containment Spray and could be thought it also causes CVI.

Technical References: OIM B-6-9a

References to be provided to applicants during exam: None

Learning Objective: 37048 - Analyze automatic features and interlocks associated with the Reactor Protection System.

- Containment Ventilation Isolation Actuation Signal

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.9

Examination Outline Cross-Reference		**Level**	RO
		Tier #	2
064 A3.01 - Ability to monitor automatic operation of the ED/G system, including: Automatic start of compressor and ED/G		**Group #**	1
		K/A #	064 A3.01
		Rating	4.1

Question 14

Unit 1 is at 100% power. All Emergency Diesel Generators (EDG) are OPERABLE.

1) If pressure is low in the "A" Starting Air Receiver for an EDG, _______ Air Compressor(s) start(s).
2) If the "A" Starting Air Receiver is completely depressurized when a complete loss of offsite power occurs, the speed indication for the associated EDG on VB4 will be _____ rpm shortly, (i.e. 1 minute) after the loss of power occurs.

A. 1) both "A" and "B"
   2) 0

B. 1) both "A" and "B"
   2) 900

C. 1) only the "A"
   2) 0

D. 1) only the "A"
   2) 900

Proposed Answer: D. 1) only the "A" 2) 900

Explanation:

A. Incorrect. The air system is normally split into A and B headers. Plausible as another air system, instrument air, is cross tied to both units and lowering pressure would affect both units. A lowering pressure in one air receiver causes only its affected compressor to start. Second part is incorrect, only one air receiver is required to start the EDG. The EDG will start (within 10 seconds) and be at rated speed, 900 rpm.
B. Incorrect. The air systems are not cross tied, only the affected train compressor starts. Either receiver can start the diesel in its required time. Second part is correct.
C. Incorrect. First part is correct, only the affected compressor starts. One train of air will start the EDG, speed will be 900 rpm (rated speed).
D. Correct. Both parts correct. Only the affected compressor starts and one train of air will start the EDG and it will reach rated speed of 900 rpm in its normal time of less than 10 seconds.

Technical References: LJ-6B, STP M-9S, Diesel Generator Operability Verification for "Starting on One Starting Train"

References to be provided to applicants during exam: None

Learning Objective: 6431 - State the purpose of Diesel Generator System components.

- Starting Air System

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.4

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
004 K6.26 Knowledge of the effect of a loss or malfunction on the following CVCS components: Methods of pressure control of solid plant (PZR relief and water inventory)	**Group #**	1
	K/A #	004 K6.26
	Rating	3.8

Question 15

GIVEN:

- Unit 1 is in MODE 5
- The RCS is solid
- RCS pressure is 350 psig
- One train of RHR in service
- PCV-135, Low Press Letdn valve, is in AUTO
- RHR Letdown Flow is being maintained via CVCS HCV-133, RHR to LTDN Flow Cont
- Charging pump 1-3 is in service

The running RHR pump trips.

As a result of the RHR pump trip, RCS pressure will _______1)_________ and CVCS Letdown pressure will ________2)__________.

A. 1) rise
   2) rise

B. 1) rise
   2) lower

C. 1) lower
   2) rise

D. 1) lower
   2) lower

Proposed Answer: B. 1) rise 2) lower

Explanation:

A. Incorrect. With the loss of letdown due to the trip of the RHR pump and continued charging flow, RCS pressure will RISE. However, PCV-135 will close down to try and maintain letdown pressure which lowers due to the RHR pump trip (valve *closes to raise* letdown pressure and *opens to lower* letdown pressure).

B. Correct. Flow into (charging) and out of (RHR/letdown) the RCS was initially balanced. When the RHR pump tripped, two things resulted: 1) the RCS pressure rose due to the flow imbalance, and 2) letdown line pressure lowered due to the loss of discharge pressure when the RHR pump tripped. PCV-135 will respond to the lowering letdown line pressure by closing to attempt to maintain pressure at setpoint.

C. Incorrect. RCS pressure rises due to charging adding inventory, while no RHR pump is running to remove inventory..When the RHR pump trips, letdown pressure lowers. PCV-135 will close down to attempt to raise pressure.

D. Incorrect. First part incorrect, RCS pressure will rise. Second part is correct..

Technical References: A-2:I , OIM B-1-1

References to be provided to applicants during exam: None

Learning Objective: Discuss Solid Plant Operations. (3356)

Question Source:	Bank #04 L161 10/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 10/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Difficulty: 3.0

Examination Outline Cross-Reference	**Level**	RO
005 A1.03 - Ability to predict and/or monitor changes in	**Tier #**	2
parameters (to prevent exceeding design limits) associated with	**Group #**	1
operating the RHRS controls including: Closed cooling water	**K/A #**	005 A1.03
flow rate and temperature	**Rating**	2.5

Question 16

GIVEN:

- Unit 2 is in MODE 6
- RHR pump 2-1 and both RHR Heat Exchangers are in service for decay heat removal
- Total RHR flow is 4000 gpm
- CCW pumps 2-1 and 2-2 are operating
- CCW pump 2-3 is out of service

If one of the running CCW pumps is lost, the operator would throttle ____1)_____ HCV-670, RHR Heat Exchanger Bypass valve, and throttle ____2)______ HCV-637/638, RHR HX Outlet valves in order to maintain RCS temperature stable.

A. 1) open
   2) shut

B. 1) open
   2) open

C. 1) shut
   2) shut

D. 1) shut
   2) open

Proposed Answer: D. 1) shut 2) open

Explanation:

A. Incorrect Throttling open HCV-670 will bypass the heat exchanger and shutting HCV-637/638 will reduce the flow through the heat exchanger resulting in a smaller cooldown rate..
B. Incorrect – Throttling open HCV-670 will bypass the heat exchanger and limit cooldown rate in addition opening HCV-670 will exceed the 5000 GPM maximum flowrate to the cold legs.
C. Incorrect. Throttling HCV-670 and HCV 637/638 shut will limit the flow through the RHR heat exchanger and reduce total flow to the RCS cold legs resulting in a reduced cooldown rate.
D. Correct. Throttling shut HCV-670 while opening HCV-637/638 will maintain the flowrate to the loops below the 5000 GPM limit in addition to increasing flow through the RHR HX to maintain the same RCS cooldown rate due to the reduced CCW flow to the RHR HX.

Technical References: OP B-2:V, RHR – Place in Service, Revision 38, LB2, Residual Heat Removal System

References to be provided to applicants during exam: None

Learning Objective: Discuss abnormal conditions associated with the RHR system (20950)

Question Source:	Bank #23 DCPP L111 11/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 11/2012	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

Difficulty: 2.8

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
103 A2.03 - Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations Phase A and B isolation	**Group #**	1
	K/A #	103 A2.03
	Rating	3.5

Question 17

GIVEN:
- A large break LOCA occurred on Unit 1
- RCS pressure is 40 psig
- Containment pressure is 25 psig and rising
- No charging pumps are running
- No RHR pumps are running
- Both SI pumps are running

For the accident in progress, the operator will _____1)_______ the RCPs because _______________2)______________________.

A. 1) trip
   2) of Phase A actuation

B. 1) trip
   2) of Phase B actuation

C. 1) NOT trip
   2) there is still cooling to the RCPs

D. 1) NOT trip
   2) there are no charging pumps running

Proposed Answer: B. 1) trip 2) of Phase B actuation.

Explanation:
Applicability: In the scenario of this question, Phase A isolation occurs on SI, and Phase B isolation occurs at 22 psig containment pressure (malfunction on containment system). Phase B isolation closes containment isolation valves associated with RCP cooling (also an "operation" of a containment system). These valves are listed in procedure STP V-11 ”Containment Isolation Phase B Valves FCV-355, FCV-356, FCV-357, FCV-363, FCV-749, and FCV-750” as meeting Technical Specification 3.6.3 (Containment Isolation Valves) SR 3.6.3.8. As such, they are included in the “Containment” Technical Specification family and meet the K/A system designator for Containment (#103). Knowledge of RCP trip criteria is RO knowledge.

A. Incorrect. First part is correct. Second part plausible because Containment isolation occurs at Phase A, but the CCW valves to the RCPs remain open and do not close until Phase B occurs.
B. Correct. Loss of RCP motor cooling occurs when Phase B occurs (22 psig). RCPs are stopped to prevent damage to the motors due to overheating.
C. Incorrect. RCPs are tripped. Plausible because CCW pumps are still available, and it could be not known the effect of high containment pressure on the CCW supply to the RCPs.
D. Incorrect because the RCPs are being secured due to losing CCW cooling. Plausible because

they would not be tripped during a LOCA if there are no “high head” pumps running. This includes the charging and/or the SI pumps. If its not known the SI pumps constitute “high head” flow (thought of as “intermediate head”), this answer is plausible

Technical References: EOP E-0 “Reactor Trip or Safety Injection”, Foldout Page; STP V-11

“Containment Isolation Phase B Valves FCV-(various)

References to be provided to applicants during exam: None

Learning Objective: State RCP trip criteria during EOP implementation. (4895)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Difficulty: 2.3

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
008 A1.02 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW temperature	**Group #**	1
	K/A #	008 A1.02
	Rating	2.9

Question 18

GIVEN:

- Unit 1 is at 100% power.
- One CCW heat exchanger is in service.
- Containment temperature is 90°F
- Spent Fuel Pool temperature is 75°F

CCW heat exchanger outlet temperature is 95°F and rising. The crew is going to reduce CCW loads in accordance with OP AP-11, Malfunction of Component Cooling Water system, Appendix B, CCW Heat Load Isolation.

In accordance with OP AP-11, Appendix B, which of the following Unit 1 CCW heat loads should the operators isolate to lower CCW temperature while still at power?

A. RCP Oil Coolers

B. Seal Water Heat Exchanger

C. Containment Fan Cooler Units

D. Spent Fuel Pool Heat Exchanger

Proposed Answer: B. Seal Water Heat Exchanger

Explanation:

A. Incorrect. The RCP coolers are isolated when the reactor is shutdown, not at power.
B. Correct. CCW isolated while the unit is at power.
C. Incorrect. According to the note in Appendix B, the CFCUs may act as a heat sink and should not be isolated
D. Incorrect. While operation would not be affected by isolating, according to the note in Appendix B, the SFP heat exchanger may act as a heat sink and should not be isolated.

Technical References: OP AP-11, Appendix B

References to be provided to applicants during exam: None

Learning Objective: 3466 - Discuss the effects and actions associated with a loss of CCW

Question Source:	Bank #33 DCPP NRC 07/2011	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 07/2011	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

Difficulty: 2.7

Examination Outline Cross-Reference	**Level**	RO

Examination Outline Cross-Reference	**Level**	RO
073 G2.2.38 – Process Radiation Monitoring: Knowledge of conditions and limitations in the facility license.	**Tier #**	2
	Group #	1
	K/A #	073 G2.2.38
	Rating	3.6

Question 19

Which of the following radiation monitors can be used to meet Technical Specifications LCO 3.4.15, RCS Leakage Detection Instrumentation?

A. RM-2, Containment Area Monitor

B. RM-11, Containment Air Particulate Monitor

C. RM-30, Containment High Range Area Monitor

D. RM-44A, Containment Exhaust Monitor

Proposed Answer: B. RM-11, Containment Air Particulate Monitor

Explanation:
Technical Specifications are part of the facility license. Above the line LCO is RO knowledge.

A. Incorrect. This is a radiation monitor in containment and plausible to think it would detect leakage into the containment atmosphere but RM-11 and 12 are the radiation monitors required by LCO 3.4.15.
B. Correct. RM-11 and RM-12 are the radiation monitors required for LCO 3.4.15
C. Incorrect. This is a radiation monitor in containment and referred to in EOPs and plausible to think it would detect leakage into the containment atmosphere but RM-11 and 12 are the radiation monitors required by LCO 3.4.15
D. Incorrect. This is a radiation monitor in containment and plausible to think it would detect leakage into the containment atmosphere and it does cause containment isolation on high radiation, but RM-11 and 12 are the radiation monitors required by LCO 3.4.15

Technical References: LG-4A, LCO 3.4.15, STP I-1A

References to be provided to applicants during exam: None

Learning Objective: Discuss significant Technical Specifications and Equipment Control Guidelines associated with the Radiation Monitoring System.

- Apply TS 3.3 and 3.4 Technical Specification LCOs. (9697C/D)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.11	

Difficulty: 2.3

Examination Outline Cross-Reference

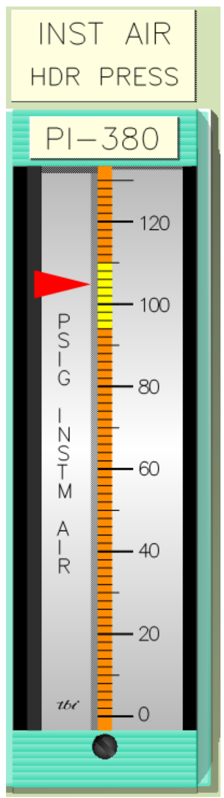
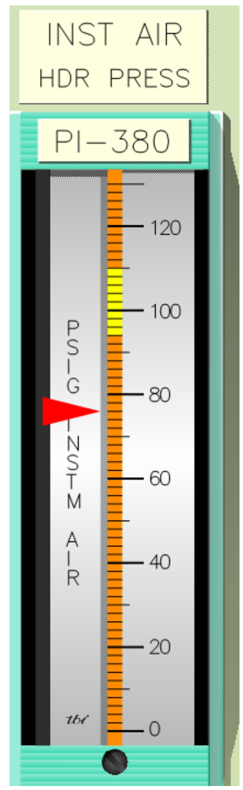
078 A4.01 - Ability to manually operate and/or monitor in the control room: Pressure gauges

Level	RO
Tier #	2
Group #	1
K/A #	078 A4.01
Rating	3.1

Question 20

Unit 1 is at 100% power.

The crew has entered OP AP-9, Loss of Instrument Air.

	
Initial Pressure	Pressure 10 minutes later

In accordance with OP AP-9, based on the PI-380 indication and rate of decrease, what will occur next?

A. Charging flow will begin to rise

B. the reciprocating air compressors will start

C. Instrument air to containment may begin to close

D. the Main Feedwater Reg valves may begin to close

Proposed Answer: D. the Main Feedwater Reg valves may begin to close

Explanation:

A. Incorrect. HCV-142 fails closed, charging will lower, not rise.
B. Incorrect. The Reciprocating air compressors start at approximately 92 psig.
C. Incorrect. FCV-584 closes at a higher pressure, approximately 85 psig.
D. Correct. According to the note in OP AP-9, the MFRV may begin to close at 75 psig.

Technical References: LB-1A, OP AP-9

References to be provided to applicants during exam: None

Learning Objective: 7209 - Discuss abnormal conditions associated with the Compressed Air System

Question Source:	Bank #54 DCPP L061C 02/2009	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 02/2009	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.0

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
022 K1.01 - Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SWS/cooling system	**Group #**	1
	K/A #	022 K1.01
	Rating	3.5

Question 21

Which of the following lists ALL Unit 2 Containment Fan Coolers cooled by CCW Vital Header A?

A. 2-1 and 2-2 only

B. 2-3 and 2-4 only

C. 2-1, 2-2 and 2-5

D. 2-3, 2-4 and 2-5

Proposed Answer: B. 2-3 and 2-4

Explanation:

A. Incorrect. 2-1 and 2-2 are two of the three on header B.
B. Correct. Only two CFCUs are cooled by header A, 2-3 and 2-4
C. Incorrect. These three are cooled by header B
D. Incorrect. 2-3 and 2-4 are cooled by header A, but 2-5 is cooled by header B.

Technical References: LH-2, OIM F-2-1

References to be provided to applicants during exam: None

Learning Objective: Describe CFCU components. • Component Cooling Water Supply (37580)

Question Source:	Bank #13 L141 04/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 04/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.8	

Difficulty: 2.0

Examination Outline Cross-Reference	**Level** RO
	Tier # 2
006 K2.04 - Knowledge of bus power supplies to the following: ESFAS-operated valves	**Group #** 1
	K/A # 006 K2.04
	Rating 3.6

Question 22

What is/are the 480 VAC power supply/supplies to the 1-1 Turbine Driven AFW Steam Generator AFW Control Valves, LCV-106, LCV-107, LCV-108 and LCV-109?

A. All four LCV's are powered from Bus G.

B. All four LCV's are powered from Bus H.

C. LCV-106 and LCV-107 are powered from Bus H. LCV-108 and LCV-109 are powered from Bus F.

D. LCV-106 and LCV-107 are powered from Bus F. LCV-108 and LCV-109 are powered from Bus G.

Proposed Answer: A. All four LCV's are powered from Bus G.

Explanation:

A. Correct. All LCV's are powered from Bus G
B. Incorrect. All are powered from the same MCC but it is G not H.
C. Incorrect. All are powered from Bus G. This is the power supply alignment for the 1-2 and 1-3 MDAFW pump LCVs.
D. Incorrect. Bus F the power supply for one the MDAFW pumps and Bus G supplies all (not half) the TDAFW valves.

Technical References: Sim VB3, LD-1

References to be provided to applicants during exam: None

Learning Objective: State the power supplies to Auxiliary Feed Water System components. (8405)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.0

Examination Outline Cross-Reference		**Level**	RO
		Tier #	2
013 K3.03 - Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Containment		**Group #**	1
		K/A #	013 K3.03
		Rating	4.3

Question 23

GIVEN:

- Unit 2 trips when a small RCS break occurs
- One reactor trip breaker remains closed
- The following PKs are LIT:
  - PK08-21, SAFETY INJECTION ACTUATION
  - PK08-22, AUTO SI BLOCKED are both lit

Subsequently, containment pressure rises from 2 to 26 psig.

1) Phase B ______________________.

2) Containment Spray ___________________________.

A. 1) actuates
   2) actuates

B. 1) actuates
   2) does not actuate

C. 1) does not actuate
   2) actuates

D. 1) does not actuate
   2) does not actuate

Proposed Answer: A. 1) actuates 2) actuates

Explanation:

A. Correct. The failure of one train of SI to reset means *one* train of CS will still have an SI signal present (PK08-20). As such, one train of spray will actuate. SI signal present is not required for Phase B, therefore, Phase B will actuate.
B. Incorrect. First part is correct. Second part incorrect. Plausible that its thought with one signal reset, both trains and therefore no train of spray will actuate
C. Incorrect. First part incorrect. SI signal does not affect Phase B. If its thought that Phase B requires the SI signal, not containment spray and one SI signal removes the signal from the logic. This is similar to de-energizing Source Range nuclear instruments, only one IR channel is needed to energize P-6 and allow de-energizing both source ranges.
D. Incorrect. Both parts incorrect. Phase B does not require SI signal. If its thought that Phase B, like Containment Spray requires SI, and one reset removes the signal, this answer is plausible.

Technical References: OIM pages B-6-5 and B-6-8

References to be provided to applicants during exam: None

Learning Objective: 37123 - Discuss abnormal conditions associated with Eagle-21/SSPS

Question Source:	Bank #10 DCPP L121 08/2014	X
(note changes; attach parent)	Modified Bank #	

	New	
	Past NRC Exam DCPP 08/2014	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 3.1

Examination Outline Cross-Reference		**Level**	RO
		Tier #	2
012 K5.02 Knowledge of the operational implications of the following concepts as the apply to the RPS: Power density		**Group #**	1
		K/A #	012 K5.02
		Rating	3.1

Question 24

What reactor trips are designed to prevent operation of the reactor with a power density of greater than 21.1 kW/foot?

A. Power Range Rate (Positive) and Power Range High Flux (High)

B. Power Range Rate (Positive) and Over Temperature Delta T (OTΔT)

C. Overpower Delta T (OPΔT) and Power Range High Flux (High)

D. Overpower Delta T (OPΔT) and Over Temperature Delta T (OTΔT)

Proposed Answer: C. Overpower Delta T (OPΔT) and Power Range High Flux (High)

Explanation:

A. Incorrect. Power Range High Flux is correct, however, PR rate (positive) is for ejected rod (flux peaking)
B. Incorrect. Power Range High Flux is correct, however, OTΔT is DNB protection.
C. Correct. OPΔT and Power Range High Flux are for excessive kw/foot (power density).
D. Incorrect. OPΔT is correct, OT is for DNB.

Technical References: OIM B-6-4-a

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the Reactor Protection System. (37048)

Question Source:	Bank #10 DCPP L162 NRC 02/2018	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 02/2018	Yes
Question History:	Last Two NRC Exams	Yes
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.2	

Difficulty: 2.5

Examination Outline Cross-Reference		**Level**	RO
		Tier #	2
010 K5.01 - Knowledge of the operational implications of the following concepts as the apply to the PZR PCS: Determination of condition of fluid in PZR, using steam tables		**Group #**	1
		K/A #	010 K5.01
		Rating	3.5

Question 25

GIVEN:
- Unit 1 Pressurizer liquid temperature is 604°F.
- Unit 1 Pressurizer vapor temperature is 618°F.
- Unit 1 RCS pressure is 1665 psig.

Given these conditions, the Unit 1 Pressurizer liquid is ____1)______ and the Pressurizer vapor is ____2)______.

A. 1) subcooled
   2) saturated

B. 1) subcooled
   2) superheated

C. 1) saturated
   2) saturated

D. 1) saturated
   2) superheated

Proposed Answer: B. 1) subcooled 2) superheated

Explanation:
Saturation temperature for 1680 psia is ~611°F (1700 psia = 613.13°F & 1650 psia = 609.05°F)
A. Incorrect. Plausible because the liquid is subcooled, however, if the steam tables are misread the vapor could be read as saturated.
B. Correct. Saturation temperature for 1665 psig is 611°F, therefore, the liquid is subcooled and the vapor is superheated.
C. Incorrect. Plausible if the steam tables are misread the liquid and vapor could be read as saturated.
D. Incorrect. Plausible because the vapor is superheated, however, if the steam tables are misread the liquid could be read as saturated.

Technical References: steam tables
References to be provided to applicants during exam: steam tables
Learning Objective: 40738 - Apply fundamentals topics associated with the Pzr, Pzr Pressure and Level Control System

Question Source:	Bank #07 L141 08/2014	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 08/2014	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	

	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.14	

Difficulty: 2.0

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
076 K1.15 Knowledge of the physical connections and/or cause-effect relationships between the SWS and the following systems: FPS	**Group #**	1
	K/A #	076 K1.15
	Rating	2.5

Question 26

The crew is performing OP AP-11, Malfunction of Component Cooling Water System, Appendix D, Instructions for Loss of Ultimate Heat Sink, following a complete loss of ASW.

In accordance with OP AP-11, Appendix D, the crew will align ____1)______ to the ____2)_____ side of the CCW Heat Exchangers.

A. 1) Fire Water
   2) tube

B. 1) Fire Water
   2) shell

C. 1) Circulating Water
   2) tube

D. 1) Circulating Water
   2) shell

Proposed Answer: A. 1) Fire Water 2) tube

Explanation:

NOTE: ASW (synonymous to SWS at DCPP), is the normal cooling system to the heat exhangers. It flows thru the tubes and CCW flows thru the shell of the heat exchangers. Firewater can be aligned as the backup source in the event of a loss of UHS.

A. Correct. The appendix aligns firewater to a CCW heat exchanger by opening drains on either end of the heat exchanger and initiating flow thru the tubes of the heat exchanger.
B. Incorrect. Second part is not correct CCW flows thru the shell of the heat exchanger.
C. Incorrect. CW is a source of water that also uses saltwater. Second part is correct.
D. Incorrect. Both parts incorrect. CW is a plausible source as both ASW and CW are saltwater sources. Shell side is plausible if its thought CCW flows thru the tubes

Technical References: LE-5, OVID 106717 sheet 8, OP AP-11 Appendix D

References to be provided to applicants during exam: None

Learning Objective: Describe ASW System components. (37013)
Describe system interrelationships between the ASW System and other plant systems. (3785)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.4	

Difficulty: 2.3

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
062 A1.01 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Significance of D/G load limits	**Group #**	1
	K/A #	062 A1.01
	Rating	3.4

Question 27

Unit 1 is at 100% power.

The crew is preparing to start and parallel Emergency Diesel Generator 1-1 to its Vital Bus in accordance with STP M-9A1, Diesel Engine Generator 1-1 Routine Surveillance Test.

In accordance with STP M-9A1, when the operator closes the output breaker, a minimum of ______1)____ MW should be picked up as soon as possible to prevent a trip due to ____________2)________________.

A. 1) 0.5
 2) overspeed

B. 1) 0.5
 2) directional (reverse) power

C. 1) 0.65
 2) overspeed

D. 1) 0.65
 2) directional (reverse) power

Proposed Answer: B. 1) 0.5 2) directional (reverse) power

Explanation:

A. Incorrect. First part is correct. Second part is incorrect. Overspeed is plausible if believed that speed will rise with the breaker closed without load.
B. Correct. STP M-9A1 states that if load is not picked up as soon as possible after closing the DG output breaker, the breaker may trip open on directional (reverse) power relay actuation and instructs the operator to raise load to 0.5 MW when the breaker is closed.
C. Incorrect. Both parts incorrect. Minimum load is 0.5 MW and trip is directional (reverse power).
D. Incorrect. First part incorrect, minimum load to pick up is 0.5 MW. 0.65 MW is an operating limit discussed in STP-9A1

Technical References: STP M-9A1, OP J-6B:IV

References to be provided to applicants during exam: None

Learning Objective: 6408 - Describe significant precautions and limitations associated with the Diesel Generator System

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.8	

Difficulty: 2.5

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
007 A2.06 - Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Bubble formation in PZR	**Group #**	1
	K/A #	007 A2.06
	Rating	2.6

Question 28

The crew is preparing to draw a bubble in the pressurizer using OP A-2:IX, Reactor Vessel-Vacuum Refill of the RCS.

In accordance with OP A-2:IX, the crew will ensure the PRT is ______1)_______ to ____________2)______________.

A. 1) filled to approximately 85%
   2) prevent air-in leakage past the PORVs and Safety Valves

B. 1) filled to approximately 85%
   2) ensure PRT gas space is purged

C. 1) drained to approximately 5%
   2) prevent air-in leakage past the PORVs and Safety Valves

D. 1) drained to approximately 5%
   2) ensure PRT gas space is purged

Proposed Answer: A. 1) filled to approximately 85%
2) prevent air-in leakage past the PORVs and Safety Valves

Explanation:

A. Correct. PRT level is raised to 85%. According to the note - PRT level is raised to approximately 85% to prevent air in-leakage past the PORVs and safety valves.
B. Incorrect. First part is correct. Second part incorrect. PRT level is raised. PRT purge is performed to reduce oxygen and hydrogen in OP A-4B:III but not for drawing a bubble, but for removing the PRT from service.
C. Incorrect. First part is incorrect. Plausible, as 3% is the level the PRT is drained to when removing from service. Second part corret. According to a note on page 13, PRT level is raised to approximately 85% to prevent air in-leakage past the PORVs and safety valves.
D. Incorrect. Both parts are incorrect.

Technical References: OP A-2:IX, OP A-4B:III, AR PK05-25

References to be provided to applicants during exam: None

Learning Objective: Draw a bubble in the Pressurizer (28370)

Question Source:	Bank #06 L161 NRC 10/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 10/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Difficulty: 3.0

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
015 K4.03 Knowledge of NIS design feature(s) and/or interlock(s) which provide for the following: Reading of source range/intermediate range/power range outside control room	**Group #**	1
	K/A #	015 K4.03
	Rating	3.9

Question 29

What nuclear instrumentation is available at the Unit 2 Hot Shutdown Panel?

1) Source Range channels N-31 and N-32
2) Gamma-Metrics neutron flux monitors, channels N53 and N54
3) Power Range channels N41, N42, N43 and N44

A. 1 only

B. 2 only

C. 1 and 3

D. 2 and 3

Proposed Answer: B. 2 only

Explanation:

N-51 Output of channel provided to:
- NI-53 (at Hot Shutdown Panel) - Source Range Indication.
- PPC Point (N0053A)
- NI-51 (at PAM-1) - Wide Range Indication

N-52 Output of channel provided to:
- NI-54 (at Hot Shutdown Panel) - Source Range Indication
- PPC Point (N0054A)
- NI-52 (at PAM-1) - Wide Range Indication
- NR-52 (at PAM-1) - Wide Range Recorder

A. Incorrect. No Excore instruments are located at the HSDP. Only gamma-metrics. Plausible as the SR channels are used to monitor power when shutdown.
B. Correct. Only gamma-metrics are on the HSDP.
C. Incorrect. No excore nuclear instruments are at the HSDP. Plausible as the SR and PR channels cover reactor power up to 100%.
D. Incorrect. Only gamma-metrics available. Plausible as this would provide power indication to 100%.

Technical References: LB-4

References to be provided to applicants during exam: None

Learning Objective: Describe controls, indications, and alarms associated with the Excore Nuclear Instrumentation System.

- Gamma-Metrics Instrumentation

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.11	

Difficulty: 2.1

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
086 K6.04 - Knowledge of the effect of a loss or malfunction on the Fire Protection System following will have on the: Fire, smoke, and heat detectors	**Group #**	2
	K/A #	086 K6.04
	Rating	2.6

Question 30

Per AR PK10-15, FIRE ALARM TROUBLE:

1) it can take as long as ____ seconds for the appropriate annunciator to scan all inputs.

2) if an event clears prior to being scanned, PK10-15 will ______________________________.

A. 1) 90
   2) not alarm

B. 1) 90
   2) still alarm

C. 1) 180
   2) not alarm

D. 1) 180
   2) still alarm

Proposed Answer: A. 1) 90 2) not alarm

Explanation:

A. Correct. According to AR PK10-15 The portion of the Fire Protection Network that originates annunciator signals **takes 90 seconds** to scan all inputs (60 seconds for IFDS). Depending on when it was last scanned, it can take up to 90 seconds (60 seconds for IFDS) for the Fire Protection Network to detect an alarming input and then alarm this annunciator. An event that clears before being scanned **will NOT** alarm.

B. Incorrect. First part is correct. Second part is incorrect. Plausible as the input could be thought to "latch" or have retentive memory such that it will still cause an alarm even its cleared. This could be thought to be a conservative fire prevention method to alert operators to a possible developing problem.

C. Incorrect. First part incorrect. 180 seconds is two times the time specified in the procedure. Second part correct.

D. Incorrect. Both parts incorrect.

Technical References: AR PK10-15

References to be provided to applicants during exam: None

Learning Objective: Describe controls, indications, and alarms associated with the Fire Detection System. (37584)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X

10CFR Part 55 Content: 55.41.7

Difficulty: 2.9

Examination Outline Cross-Reference		**Level**	RO
		Tier #	2
033 A1.01 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Spent Fuel Pool Cooling System controls including: Spent fuel pool water level		**Group #**	2
		K/A #	033 A1.01
		Rating	2.7

Question 31

GIVEN:

- Unit 2 is 100% power
- PK11-04, SPENT FUEL POOL LVL/TEMP alarms due to input 1061, Spent Fuel Pool Lvl Lo
- Spent Fuel Pool level indicators, LI-801 and LI-802, indicate Spent Fuel Pool level is 24' and lowering slowly

The cause for the lowering level has not been identified.

In accordance with OP AP-22, Spent Fuel Pool Abnormalities, Appendix A, Addition of Water to the Spent Fuel Pool, for the current plant conditions, what is the preferred method of makeup to the Spent Fuel Pool?

A. The CST

B. The RWST

C. Ionics system

D. 4% Boric Acid prepared BART

Proposed Answer: B. The RWST

Explanation:

A. Incorrect. The CST is a source of makeup to the SFP when the leakage is evaporation (after Ionics and the Transfer Tank).
B. Correct. The RWST is the preferred source of water for leakage, ISFSI MPC loading or for unknown reasons.
C. Incorrect. Ionics is the preferred source for boiling or evaporation.
D. Incorrect. The BART is a source of borated water that could be used but is not the preferred borated source. Plausible because it would be used ahead of some other sources such as the Refueling Cavity.

Technical References: AR PK11-04, OP AP-22 Appendix A

References to be provided to applicants during exam: None

Learning Objective: Discuss abnormal conditions associated with the Spent Fuel Pool Cooling System, (40509)

Question Source:	Bank #61 L091 07/2011	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 07/2011	Yes
Question History:	Last Two NRC Exams	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Difficulty: 3.0

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
041 A2.02 -Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations Steam valve stuck open	**Group #**	2
	K/A #	041 A2.02
	Rating	3.6

Question 32

GIVEN:

- The crew has entered EOP E-0.1, Reactor Trip Response, following a reactor trip on Unit 1
- All RCPs are running
- AFW flow is 450 gpm
- Steam Generator Narrow Range levels are off scale low
- MSRs have been RESET
- Blowdown is isolated

The Unit 1 CO notes that RCS temperature is 540ºF and lowering and red and green lights are lit for two Group 2 steam dump valves. Only green lights are lit for all other steam dump valves.

In accordance with EOP E-0.1, the operator will:

A. close the MSIVs and MSIV Bypass valves.

B. place the STEAM DUMP CONTROL BYPASS SELECT switches, 43/SDA and 43/SDB in OFF RESET.

C. continue to monitor the steam dumps and check that they close when P-12 actuates.

D. place 40% STM DUMP VLVS PRESS CONT, HC-507 to MANUAL and press the DEC pushbutton.

Proposed Answer: A. close the MSIVs and MSIV Bypass valves.

Explanation:

A. Correct. Step 1.b. checks RCS temperature stable or trending to 547°F. If temperature is less than 547°F and lowering, actions are taken to stop/control the cooldown. With AFW flow already reduced and MSRs closed, the only action left is to close the MSIVs and bypass valves.

B. Incorrect. Placing the steam dumps in OFF will not cause the 2 valves that are open to close – because temperature is less than P-12 already, all steam dumps have a closed signal at this time.

C. Incorrect. Temperature is already below P-12. Plausible if the P-12 setpoint is not known.

D. Incorrect. Placing steam dumps in MANUAL would not close the Group 2 valves. Below P-11, the controller will not affect single steam dump operation.

Technical References: EOP E-0.1

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #62 DCPP NRC L031	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam #62 DCPP NRC 02/2005	Yes

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Difficulty: 3.1

Examination Outline Cross-Reference		**Level**	RO
		Tier #	2
068 A3.02 - Ability to monitor automatic operation of the		**Group #**	2
		K/A #	068 A3.02
Liquid Radwaste System including Automatic isolation		**Rating**	3.6

Question 33

Liquid Radwaste radiation monitor, RE-18 has just exceeded its setpoint and stopped a release.

What automatic action(s) occurred?

NOTE:

1. RCV-18, Liquid Waste to Overboard, closed
2. FCV-647, Filter 0-4 to ASW Overboard or EDRs, closed
3. FCV-477, Filters 04 and 05 outlet to EDRs, opened

A. 1 only

B. 2 only

C. 1 and 3

D. 2 and 3

Proposed Answer: C. 1 and 3

Explanation:

A. Incorrect. RCV-18 closes but additionally, FCV-477 opens.
B. Incorrect. FCV-647 does not close. Plausible as it is the next valve downstream of RE-18.
C. Correct. RE-18 in high alarm causes isolation by closing RCV-18 and opening a recirc path back by opening FCV-477
D. Incorrect. FCV-477 opens, however, RCV-18, not FCV-647 closes.

Technical References: OIM G-1-1 and G-3-1

References to be provided to applicants during exam: None

Learning Objective: 69251 - Explain the automatic actions associated with the Liquid Radwaste system.

Question Source:	Bank #61 DCPP L051 04/2007	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam #61 DCPP NRC 04/2007	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.11	

Difficulty: 2.2

Examination Outline Cross-Reference		**Level**	RO
		Tier #	2
027 A4.03 - Ability to manually operate and/or monitor in the control room: CIRS fans		**Group #**	2
		K/A #	027 A4.03
		Rating	3.3

Question 34

1) During normal 100% power operations, ________ of the Containment Iodine Removal Fans are normally in service.
2) In EOP E-0, Reactor Trip of Safety Injection, Appendix E, ESF Auto Actions, Secondary and Auxiliaries Status, the operator ensures the Containment Iodine Removal Fans are ____________________.

A. 1) none
   2) ON

B. 1) none
   2) OFF

C. 1) one
   2) ON

D. 1) one
   2) OFF

Proposed Answer: B. 1) none 2) OFF

Explanation:

A. Incorrect. First part is correct. CIR fans are run to cleanup containment atmosphere when requested by chemistry. Normally, in OP L-1, "Plant Heat Up from Hot Shutdown to Hot Standby," OP L-5, "Plant Cooldown from Minimum Load to Cold Shutdown," or for any scheduled entry into Containment when iodine level is unacceptably high.
B. Correct. At power, the CIR fans are not run unless containment atmosphere cleanup is necessary before containment entry. During an accident, because the charcoal filters may catch fire, E-0 appendix E ensures the fans are off.
C. Incorrect. Both parts are incorrect. No units are normally in service and they are not run during accidents.
D. Incorrect. First part is incorrect. Second part is correct.

Technical References: E-0 Appendix E, LH-3

References to be provided to applicants during exam: None

Learning Objective: Discuss significant precautions and limitations associated with the Iodine Removal System. (5240)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.0

Examination Outline Cross-Reference

028 G2.1.27 – Hydrogen Recombiner and Purge Control – Knowledge of system purpose and/or function.

Level	RO
Tier #	2
Group #	2
K/A #	028 G2.1.27
Rating	3.9

Question 35

GIVEN:

- Large Break LOCA is in progress
- Core damage is occurring

Which of the following describes the minimum equipment that is needed in operation to maintain hydrogen at or below limits?

A. One Hydrogen Recombiner only

B. Both Hydrogen Recombiners

C. One train of Containment Spray only

D. One train of Containment Spray and one Hydrogen Recombiner

Proposed Answer: A. One Hydrogen Recombiner only

Explanation:

A. Correct. Only one recombiner is required. Each has 100% capacity.
B. Incorrect. More than the minimum, only one required.
C. Incorrect. Only train of Containment Spray is required to maintain containment pressure below design but is not credited for hydrogen removal. Plausible if iodine removal is confused with hydrogen removal.
D. Incorrect. Containment Spray is not credited for hydrogen removal.

Technical References: OP H-9, LI-2

References to be provided to applicants during exam: None

Learning Objective: 40834 - Explain significant CHPS design features and the importance to nuclear safety

Question Source:	Bank #33 L121 08/2014	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 08/2014	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.8	

Difficulty: 2.8

Examination Outline Cross-Reference		**Level**	RO
		Tier #	2
045 K1.18 - Knowledge of the physical connections and/or cause-effect relationships between the MT/G system and the following systems: RPS		**Group #**	2
		K/A #	045 K1.18
		Rating	3.6

Question 36

1) P-9, Power Range Permissive, is enabled to cause a reactor trip when Reactor power is above __________.
2) In EOP E-0, Reactor Trip or Safety Injection, a turbine trip is verified by ensuring a minimum of _________ Stop Valves Closed.

A. 1) 35%
   2) 2

B. 1) 35%
   2) 4

C. 1) 50%
   2) 2

D. 1) 50%
   2) 4

Proposed Answer: D. 1) 50% 2) 4

Explanation:

A. Incorrect. Both parts incorrect. A turbine trip will cause a reactor trip above 50% (P-9). The trip signal is 4 of 4 stop valves closed, which is what is checked in E-0 as part of the immediate actions to verify the turbine is tripped. Both plausible, 35% is the setpoint for P-8, Loss of Flow Permissive (RCS loop flow trip shift from 2 of 4 to 1 of 4). 2 of 4 is the coincidence for many reactor trips.
B. Incorrect. First part incorrect, second part correct. 4 stop valves closed is checked as part of the immediate actions in E-0.
C. Incorrect. First part correct. Second part incorrect, all 4 stop valves are checked, not 2.
D. Correct. First part correct, above P-9, 50%, a turbine trip will cause a reactor trip. Second part correct, all stop valves (4) are checked in E-0.

Technical References: OIM B-6-2, EOP E-0

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the Reactor Protection System. • Turbine Trip(37048)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 3.3

Examination Outline Cross-Reference	**Level**	RO
	Tier #	2
014 K5.01 - Knowledge of the operational implications of the following concepts as they apply to the RPIS: Reasons for differences between RPIS and step counter	**Group #**	2
	K/A #	014 K5.01
	Rating	2.7

Question 37

Of the two Rod Position Indication Systems:

1) The most accurate is ______________________________.

2) The most reliable is _______________________________.

NOTE:
- DRPI – Digital Rod Position Indication
- BDPI – Bank Demand Position Indication

A. 1) DRPI
   2) BDPI

B. 1) DRPI
   2) DRPI

C. 1) BDPI
   2) DRPI

D. 1) BDPI
   2) BDPI

Proposed Answer: C. 1) BDPI 2) DRPI

Explanation:
To answer the question, it must be known how each system operates that makes each one the most reliable or accurate. Accuracy: The BDPI system is more accurate, capable of an accuracy of being within 1 step (5/8 inch) of the actual rod position. This is much more accurate than the Digital Rod Position Indication (DRPI) which has an accuracy of +4 steps. BDPI is considered to not be very reliable since it infers rod position indication without actually measuring it. In comparison, DRPI actually measures rod position and is thus more reliable.
A. Incorrect. Answers are backwards. DRPI more reliable, BDPI more accurate
B. Incorrect. While DRPI is the most reliable, (second part correct), the most accurate is BDPI.
C. Correct. BDPI is more accurate while DRPI is more reliable.
D. Incorrect. DRPI is more reliable.

Technical References: LA-3A

References to be provided to applicants during exam: None

Learning Objective: Describe Rod Control System components. (36987)

Question Source:	Bank #35 DCPP NRC L121 08/2014	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 08/2014	Yes
Question History:	Last Two NRC Exams	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.2	

Difficulty: 2.0

Examination Outline Cross-Reference		**Level**	RO
		Tier #	2
017 K3.01 - Knowledge of the effect that a loss or malfunction of the ITM system will have on the following: Natural circulation indications		**Group #**	2
		K/A #	017 K3.01
		Rating	3.5

Question 38

The crew is verifying natural circulation is established in accordance with E-0.1, Reactor Trip Response. Subcooling, as displayed on VB2, is 40°F. All core exit thermocouples are reading 570°F.

A single core exit thermocouple failure:

A. high or low, will affect subcooling indication.

B. high or low, will not affect subcooling indication.

C. low will affect indication; high failure will not affect indication.

D. high will affect indication; low failure will not affect indication.

Proposed Answer: D. high will affect indication; low failure will not affect indication.

Explanation:

A. Incorrect. Unlike wide range pressure, which is used to determine train specific subcooling, only the highest temperature is used. A low failure would not affect indication.
B. Incorrect. With PCS components, once a parameter is out of range, it is automatically deselected. Subcooling determination is not part of the PCS instrumentation.
C. Incorrect. Subcooling uses the highest temperature.
D. Correct. The highest temperature, Thot or thermocouple is fed to the Subcooled Margin Monitor for determining subcooling.

Technical References: LA-2D, E-0.1

References to be provided to applicants during exam: None

Learning Objective: Describe RVLIS components • Subcooled Margin Monitor

Question Source:	Bank #33 DCPP L141 04/2016	X
(note changes; attach parent)	Modified Bank	
	New	
	Past NRC Exam DCPP NRC 04/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.2	

Difficulty: 2.3

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
EPE 055 EK3.02 - Knowledge of the reasons for the following responses as the apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power		**Group #**	1
		K/A #	EPE 055 EK3.02
		Rating	4.3

Question 39

The crew is performing EOP ECA-0.0, Loss of Vital AC Power, step 18, DEPRESSURIZE Intact Steam Generators To Reduce RCS Pressure To Inject Accumulators.

According to the background document for EOP ECA-0.0, the reason for stopping the depressurization when steam generator pressure is less than 300 psig is to prevent:

A. injection of accumulator nitrogen.

B. drawing a bubble in the reactor vessel head.

C. tube failure due to high RCS/steam generator DP.

D. challenging RCS Integrity Critical Safety Function.

Proposed Answer: A. injection of accumulator nitrogen.

Explanation:

A. Correct. Per ECA-0.0 background, the target SG pressure for Step 18 should ensure that RCS pressure is above the minimum pressure to preclude injection of accumulator nitrogen into the RCS. The target SG pressure should be based on the nominal SG pressure to preclude nitrogen addition, plus margin for controllability (e.g., 100 psi).
B. Incorrect. During the depressurization a head bubble may occur (caution states the depressurization is not stopped if it occurs). Plausible as most procedures try to avoid bubble formation.
C. Incorrect. DP across steam generator U-tubes has a normal 1700 psid limit. This is not the concern during the depressurization.
D. Incorrect. This is the reason for stopping at 310°F.

Technical References: ECA-0.0 and background

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #DCPP Bank B-1021	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Difficulty: 2.0

Examination Outline Cross-Reference	**Level**	RO
	Tier #	1
EPE 007 EA1.03 - Ability to operate and monitor the following as they apply to a reactor trip: RCS pressure and temperature	**Group #**	1
	K/A #	EPE 007 EA1.03
	Rating	4.2

Question 40

The crew is performing EOP E-0, Reactor Trip or Safety Injection. RCS temperature is 542°F and rising slowly.

In EOP E-0,

1) per the Foldout page, the operator will trip the RCPs if RCS ________________________.
2) at Step 6, CHECK RCS Temperature, the operator will stabilize temperature at ________________________________.

A. 1) subcooling is less than 20°F
   2) 542°F

B. 1) subcooling is less than 20°F
   2) 547°F

C. 1) pressure is less than 1300 psig
   2) 542°F

D. 1) pressure is less than 1300 psig
   2) 547°F

Proposed Answer: C. 1) pressure is less than 1300 psig 2) 542°F

Explanation:

A. Incorrect. Subcooling is used for many EOP transitions, SI termination, and RCP trip criteria, however, RCS pressure is the criteria used for RCP trip. Second part correct.
B. Incorrect. First part incorrect.. Second part is incorrect. If temperature is higher, action is taken to lower temperature to 547°F. Also, no load RCS temperature is 547°F and steps in the EOPs check is temperature is trending to 547°F.
C. Correct. Both parts correct. RCS pressure less than 1300 psig is the criteria for RCP trip. If temperature is low, action is taken to stabilize temperature at its current value to limit the inventory in the pressurizer in the event of an inadvertent SI.
D. Incorrect. First is correct. Second part is incorrect.

Technical References: EOP E-0

References to be provided to applicants during exam: None

Learning Objective: 4895 - State RCP trip criteria during EOP implementation
<u>77210 - From memory, explain the guidance to respond to a reactor trip, per EOP E-0 and EOP E-0.1.</u>

Qquestion uestion Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Difficulty: 2.3

Examination Outline Cross-Reference	**Level**	RO
	Tier #	1
APE 040 AA2.01 - Ability to determine and interpret the following as they apply to the Steam Line Rupture: Occurrence and location of a steam line rupture from pressure and flow indications	**Group #**	1
	K/A #	APE 040 AA2.01
	Rating	4.2

Question 41

Unit 1 is at 15% power.

A steam break occurs causing:
- Reactor trip
- SI
- MSI
- FWI
- All equipment operated as designed

Current Steam Generator Pressures:
- 1-1 – 900 psig, stable
- 1-2 – completely depressurized
- 1-3 – completely depressurized
- 1-4 – 870 psig, stable

The steam break:

A. cannot be isolated.

B. was isolated when the MSIVs closed.

C. will be isolated by the operator closing TDAFW pump 1-1 Steam Inlet valve, FCV-95.

D. will be isolated by the operator closing both TDAFW pump 1-1 Steam Supply valves, FCV-37 and FCV-38.

Proposed Answer: D. will be isolated by the operator closing both TDAFW pump 1-1 Steam Supply valves, FCV-37 and FCV-38

Explanation:

A. Incorrect. The break is downstream of the steam valves to the TDAFW and can be isolated. Plausible if its thought the break is upstream of the steam headers to the TDAFW pump. If so, the break would be unisolable.

B. Incorrect. The break is not isolated. Plausible if its thought the pressure is lowering in the 1-2 and 1-3 steam generators due to the TDAFW pump running.

C. Incorrect. FCV-95 is closed at power. If there was a leak downstream of the valve, the plant would not have tripped.

D. Correct. The break is downstream of the normally open FCV-37 and 38. Closing both valves will isolate the leak.

Technical References: OVID 106704 sheet 4

References to be provided to applicants during exam: None

Learning Objective: 40561 - Describe the basic flow path of the main steam system

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #48 DCPP L051 04/2007	X
	New	
	Past NRC Exam DCPP NRC 04/2007	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 3.2

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
APE 008 G2.4.4 – Pressurizer Vapor Space Accident: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.		**Group #**	1
		K/A #	APE 008 G2.4.4
		Rating	4.5

Question 42

GIVEN:
- Unit 1 is at 100% power
- RCS pressure is 2235 psig and lowering slowly
- PRT pressure is 5 psig and rising slowly

For the current plant conditions, what indication(s) would cause the crew to confirm there is seat leakage through a PORV for entry into OP AP-1, Excessive Reactor Coolant System Leakage?

1. PORV tailpipe temperature indication, TI-463, reads top of scale, 400°F
2. PORV tailpipe temperature indication, TI-463, reads 230°F and rising
3. Sonic Flow is indicated

A. 1 only

B. 2 only

C. 1 and 3

D. 2 and 3

Proposed Answer: B. 2 only

Explanation:

A. Incorrect. A reading at top of scale is most likely an instrument failure. Also, saturation temperature for 400°F is approximately 225 psig, well above the PRT pressure. Plausible if its thought temperature is driven by pressurizer temperature, 650°F.

B. Correct. The isenthalpic process would have tailpipe temperature going to saturation temperature for the PRT. At 5 psig, (20 psia), this temperature is approximately 230°F. Also, the sonic indicators, unlike the tailpipe temperatures, are on separate headers for each pressurizer safety and do not respond to steam leaking past a PORV, only safety steam flow.

C. Incorrect. Both parts are incorrect.

D. Incorrect. First part is correct. However, sonic flow will not respond to PORV leakby, only Pressurizer Safety leakby.

Technical References: AR PK05-23, steam tables, LPA-13

References to be provided to applicants during exam: Steam Tables

Learning Objective: Discuss why having a solid understanding of plant design, engineering principles, and sciences is a necessary operator fundamental. (56220)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

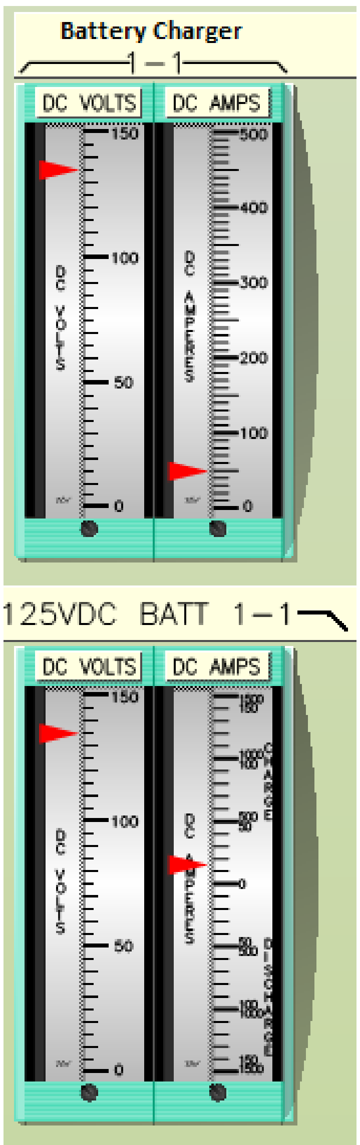
Difficulty: 2.2

Examination Outline Cross-Reference	**Level**	RO
	Tier #	1
APE 058 AK1.01 - Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation	**Group #**	1
	K/A #	APE 058 AK1.01
	Rating	2.8

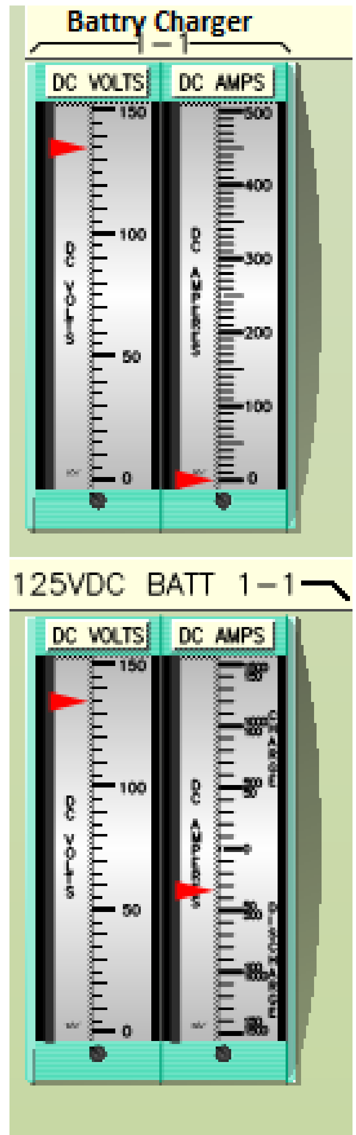
Question 43

Unit 1 is at 100% power.

Initial Indications



Current Indications



The current indications are consistent with:

A. the loss of DC Bus 1-1.

B. a loss of 480 VAC bus H.

C. placing of the battery on equalizing charge.

D. opening the battery charger output breaker.

Proposed Answer: D. opening the battery charger output breaker.

Explanation:

A. Incorrect. While DC amps of the charger are at 0, negative amps on the battery indicate the battery is carrying the bus, not a loss of the bus.
B. Incorrect. The normal supply to Battery Charger 1-1 is bus F. loss of Bus H would not impact DC bus 1-1. Plausible as EDG 1-1 supplies bus H.
C. Incorrect. Equalizing charge would have higher battery voltage and there would still be amps indicated on the charger.
D. Correct. Opening the charger output breaker would result in the battery supplying the bus. Indications would be 0 amps from the charger and negative amps from the battery, as it is now carrying load. .

Technical References: OIM J-1-1 and J-1-2

References to be provided to applicants during exam: None

Learning Objective: Discuss abnormal conditions associated with the DC Power System. (5193)

Question Source:	Bank #49 DCPP L091 07/2011	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam #49 DCPP NRC 07/2011	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.5

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
APE 025 AK2.01 - Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: RHR heat exchangers		**Group #**	1
		K/A #	APE 025 AK2.01
		Rating	2.9

Question 44

GIVEN:
- Unit 1 is in MODE 5
- Both trains of RHR are in service
- Total RHR flow is 3000 gpm (both RHR pumps running)

Instrument air pressure to HCV-637, 1-2 RHR Heat Exchanger outlet valve, has just been lost.

RHR flow to loop 3 and 4 cold legs will:

A. lower to zero.

B. lower to a minimum flow limited by a mechanical stop.

C. rise to runout conditions.

D. rise to a maximum flow limited by a mechanical stop.

Proposed Answer: D. rise to a maximum flow limited by a mechanical stop

Explanation:
A. Incorrect HCV-637 will fail open raising flow. Plausible because it may be thought that the valve will fail closed on loss of air as there is not a mechanical stop on closing.
B. Incorrect. HCV-637 will fail open raising flow. Plausible because there is a mechanical stop to ensure some minimum flow (like there is for opening), then this is a logical answer.
C. Incorrect. HCV-637 will fail open raising flow. However, it is limited by a stop to prevent runout. If this is not known, runout is a logical answer.
D. Correct. RHR heat exchanger outlet valves (HCV-637 and 638) are fail open valves. With HCV-637 open (to its stop) flow can rise to between 3976 gpm and < 4319 gpm, limited by a mechanical stop. Flow to the loops will rise.

Technical References: LB-2; OIM B-3-1

References to be provided to applicants during exam: None

Learning Objective: Discuss abnormal conditions associated with the RHR system (20950)

Question Source:	Bank #04 DCPP L141 04/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 04/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

Difficulty: 2.3

Examination Outline Cross-Reference	**Level**	RO
	Tier #	1
APE 022 AK3.06 - Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: RCP thermal barrier cooling	**Group #**	1
	K/A #	APE 022 AK3.06
	Rating	3.2

Question 45

Unit 1 is at 100% power.

A loss of seal injection to all RCPs occurs.

In accordance with OP AP-28, Reactor Coolant Pump Malfunction, stopping of all RCPs ___1)___ required ______________2)______________.

A. 1) is
   2) within 5 minutes

B. 1) is
   2) within 15 minutes

C. 1) is NOT
   2) because RCP seal flow is now from CCW

D. 1) is NOT
   2) because RCP seal flow is now from RCS cooled by the thermal barrier heat exchanger

Proposed Answer: D. 1) is NOT 2) because RCP seal flow is now from RCS cooled by the thermal barrier heat exchanger

Explanation:

A. Incorrect. A trip is not required as long as CCW supplies the RCP thermal barrier. Plausible as SDS (passive Shutdown Seal) actuation is a concern if all cooling is lost and the seal heats up to greater than 260°F and the RCPs are tripped within 5 minutes of a complete loss of CCW cooling.
B. Incorrect. A trip is not required as long as CCW supplies the RCP thermal barrier. Plausible as 15 minutes is used as a time for actions such as making E-plan classifications or during a reactor start up prior to withdrawing control rods.
C. Incorrect. First part is correct. CCW cools the thermal barrier heat exchanger, it does not replace the charging injection. Plausible if the flow paths are not understood.
D. Correct. The loss of seal injection allows RCS to begin to flow up the shaft of the RCP. This RCS is cooled to approximately 140°F as it travels up the shaft of the pump and enters the number 1 seal. This is acceptable as long as CCW is supplying the thermal barrier.

Technical References: LA-6, OP L-1, AR PK04-22
References to be provided to applicants during exam: None
Learning Objective: Discuss abnormal conditions associated with the RCP. (35744)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.3	

Difficulty: 2.0

Examination Outline Cross-Reference	Level	RO
	Tier #	1
APE 027 AA1.01 - Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: PZR heaters, sprays, and PORVs	Group #	1
	K/A #	APE 027 AA1.01
	Rating	4.0

Question 46

Unit 1 is at 100% power.

Following a transient, HC-455K, PZR Press Control, output is 90.0% and rising.

1) Pressurizer Spray valves will be _____________.
2) Pressurizer PORV(s) will be _____________.

A. 1) partially open
   2) closed

B. 1) partially open
   2) open

C. 1) fully open
   2) closed

D. 1) fully open
   2) open

Proposed Answer: D. 1) fully open 2) open

Explanation:

A. Incorrect. Based on the current output, the sprays and PORVs will be open. The spray valves begin to open at a setpoint of 40.6% which corresponds to a pressure 25 psig above 2235 (2260). The spray valves are fully open at 71.9% or +75 psig (2310 psig). The PORVs open at 87.5% or 100 psig above setpoint (2335 psig) If the fully open spray valve setpoint is not known, its plausible to think only pressurizer spray valves are open.
B. Incorrect. Plausible if its thought the setpoints for less than fully open sprays and PORVs open overlap.
C. Incorrect. There is a pressure band when the sprays are open and PORVs closed, but this is from 71.9% to 87.5% (2310 to 2335 psig)
D. Correct. Sprays are fully open at 71.9% and PORVs open at 87.8%. If output is 90%, sprays are fully open and the PORVs are also open.

Technical References: OIM page A-4-5

References to be provided to applicants during exam: None

Learning Objective: Describe the operation of the Pressurizer, Pressure & Level Control System. (4585)

Question Source:	Bank # 58 2016 South Texas Project,	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam 2016 STP NRC Exam	Yes
Question History:	Last Two NRC Exams	No

Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.6

Examination Outline Cross-Reference	**Level**	RO
	Tier #	1
E05 EA2.2 - Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	**Group #**	1
	K/A #	E05 EA2.2
	Rating	3.7

Question 47

GIVEN:

- A total loss of Main Feedwater and Auxiliary Feedwater occurs on Unit 1
- The crew is performing the actions of EOP FR-H.1, Response to Loss of Secondary Heat Sink
- Bleed and Feed has been initiated
- All steam generators are "dry"
- Core Exit Thermocouples are 555°F, rising slowly

The capability to feed all steam generators using the TDAFW pump has been restored.

Per EOP FR-H.1, what action should be taken by the crew?

A. Fully open <u>one</u> TDAFW LCV to establish maximum AFW flow to one steam generator.

B. Fully open <u>all</u> TDAFW LCVs to establish maximum AFW flow to all steam generators.

C. Throttle open <u>one</u> TDAFW LCV to establish AFW flow of 25 to 100 gpm to one steam generator.

D. Throttle open <u>all</u> TDAFW LCVs to establish AFW flow of 25 to 100 gpm to each steam generator.

Proposed Answer: A. Fully open <u>one</u> TDAFW LCV to establish maximum AFW flow to one steam generator.

Explanation:

A. Correct. With core exit temperatures increasing, maximum flow to restore a heat sink as quickly as possible is necessary.
B. Incorrect because only one steam generator is used in order to limit potential faults to only that steam generator. Plausible because the need to re-establish a heat sink is urgent and the TDAFW pump would normally be used to feed all steam generators.
C. Incorrect because maximum flow to one steam generator is to be used if core exit thermocouples are rising. Plausible because 100 gpm is the top end of the allowable flow band (25-100 gpm) to one dry steam generator if core exit temperatures are lowering.
D. Incorrect because only one steam generator is used in order to limit potential faults to only that steam generator. Plausible because the TDAFW pump would normally be used to feed all steam generators, and the restricted flow band of 25-100 gpm would be used if core exit temperatures were stable or lowering.

Technical References: EOP FR-H.1 Foldout page

References to be provided to applicants during exam: None

Learning Objective: Explain feeding a dry S/G including: (6375)

• Definition of dry S/G

• Effects of feeding a dry S/G

Question Source:	Bank #50 L161 10/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 10/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

Difficulty: 2.7

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
E11 G2.2.22 – Loss of Emergency Coolant Recirc - Knowledge of limiting conditions for operations and safety limits.		**Group #**	1
		K/A #	E11 G2.2.22
		Rating	4.0

Question 48

The crew has entered EOP ECA-1.1, Loss of Emergency Recirculation.

They are performing step 13, "ESTABLISH One Train Of SI Flow".

In accordance with EOP ECA-1.1, which of the following alignments for Charging Pumps (CCP) and Safety Injection Pumps (SIP) would be acceptable?

A. CCP 1-1 and SIP 1-1

B. CCP 1-2 and SIP 1-2

C. CCP 1-3 and SIP 1-1

D. CCP 1-3 and SIP 1-2

Proposed Answer: B. CCP 1-2 and SIP 1-2

Explanation:

NOTE: bank question KA is EA2.2 which has wording that mirrors selected generic KA.

The limiting condition for operation is the application of the note which states: The ECCS CCPs and SI Pps should be stopped in alternate trains when possible. According to the DCPP background document the basis is so if one train is disabled, some flow will still be available.

A. Incorrect. The note before step 13 provides guidance to leave pumps in alternate trains. running. This refers to power supplies such that if a single power supply was lost, that all running equipment would not be simultaneously lost. CCP 1-1 and SIP 1-1 are both from bus F and does not meet the alternate train guidance.
B. Correct. CCP 1-2 and SIP 1-2 meet the alternate train requirement because they are powered from Vital 4 kV buses G and H respectively.
C. Incorrect. Either CCP 1-1 or 1-2 should be running. CCP 1-3 is the non-ECCS CCP. Plausible because CCP 1-3 is powered from bus G (train B) and SIP 1-1 is from bus F (train A).
D. Incorrect. Either CCP 1-1 or 1-2 should be running during accident conditions, although CCP 1-3 is functional, it is the non-ECCS CCP. Plausible because CCP 1-3 and SIP 1-2 are powered from different buses, which meets the guidance in the note prior to step 13.

Technical References: EOP ECA-1.1 and background

References to be provided to applicants during exam: None

Learning Objective: Explain basis of emergency procedure steps for ECA-1.1. (42460)

Question Source:	Bank #55 DCPP L141 04/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 04/2016	Yes

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Difficulty: 3.1

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
APE 062 AK3.03 - Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: Guidance actions contained in EOP for Loss of nuclear service water		**Group #**	1
		K/A #	APE 062 AK3.03
		Rating	4.0

Question 49

The crew is aligning Unit 1 for Cold Leg Recirculation in accordance with EOP E-1.3, Transfer to Cold Leg Recirculation.

Only one train of ASW is available.

When the crew completes the alignment, the reason only one RHR heat exchanger and three CFCUs will be in operation is to prevent:

A. runout of the ASW pump.

B. runout of the CCW pump.

C. flashing and water hammer in the ASW system.

D. exceeding CCW system temperature design limit.

Proposed Answer: D. exceeding CCW system temperature design limit.

Explanation:
Design of the ASW (Diablo equivalent to Nuclear Service Water) system is to adequately remove CCW heat. EOP E-1.3 includes operator actions to limit the heat loads during post-LOCA cold-leg recirculation if less than two ASW pumps and two CCW heat exchangers are in Service (a loss of one train).

A. Incorrect. With a single train of CCW in service, its plausible the student could focus on the single CCW train and believe runout is a possibility. The problem is due to the lack of cooling (from ASW), CCW heat removal is reduced to the point that if loads are not restricted, design temperature could be exceeded.
B. Incorrect. The system is designed for one pump to supply both CCW trains during normal operation without runout. Plausible because, if one train is supplying both trains of CCW, then its possible to think the loads could cause the only running ASW pump to approach runout.
C. Incorrect. Temperature remains less than saturation. Plausible because if two trains of CCW were put into service with only one train of ASW, temperature will rise.
D. Correct. The heat load of all the loads on the CCW with only one ASW train could cause the system to not be able to meet its purpose to remove heat from the CCW system and the CCW system could exceed its design temperature

Technical References: EOP E-1.3, LF-2, FSAR

References to be provided to applicants during exam: None

Learning Objective: 8105 - Explain significant CCW system design features and the importance to nuclear safety

Question Source: Bank #53 DCPP L162 02/2018 X

(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 02/2018	Yes
Question History:	Last Two NRC Exams	Yes
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.8	

Difficulty: 2.2

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
E04 EK2.2 - Knowledge of the interrelations between the (LOCA Outside Containment) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.		**Group #**	1
		K/A #	E04 EK2.2
		Rating	3.8

Question 50

The crew is performing the actions in EOP ECA-1.2, LOCA Outside Containment.

In accordance with EOP ECA-1.2:

1) what system is isolated when attempting to locate the leak?

2) what indication is used to determine if the leak has been isolated?

A. 1) RHR
   2) Pressurizer level

B. 1) RHR
   2) RCS pressure

C. 1) CCW
   2) Pressurizer level

D. 1) CCW
   2) RCS pressure

Proposed Answer: B. 1) RHR 2) RCS pressure

Explanation:

A. Incorrect. Incorrect. Pressurizer level will increase, but after pressure begins to rise and the system is refilled. RCS pressure is checked in ECA-1.2
B. Correct. RHR is isolated, one train at a time. If the leak is isolated, pressure will respond and quickly repressurize the system as SI flows into the now intact system.
C. Incorrect. Both parts incorrect. RHR is isolated, as it is the low pressure system and deemed to be the most likely location for a LOCA outside containment. CCW plausible as it is also a low pressure system that is connected to the RCS and RHR. The procedure checks RCS pressure not pressurizer level.
D. Incorrect. Second part is correct, however, RHR not CCW is isolated. CCW plausible as it is also a low pressure system that is connected to the RCS and RHR.

Technical References: ECA-1.2

References to be provided to applicants during exam: None

Learning Objective: Explain basis of emergency procedure steps for ECA-1.2. (42461, 7920H)

Question Source:	Bank #53 L091C 03/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 03/3012	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	

	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.5	

Difficulty: 2.6

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
APE 057 AK3.01 - Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus		**Group #**	1
		K/A #	APE 057 AK3.01
		Rating	4.1

Question 51

Unit 1 is at 100% power.

All instrument AC power from PY-13 and PY-14 is lost.

1) __________ reactor trip breaker(s) will be open.

2) PK08-21, SAFETY INJECTION ACTUATION will be ___________.

A. 1) Only one
   2) lit

B. 1) Only one
   2) NOT lit

C. 1) Both
   2) lit

D. 1) Both
   2) NOT lit

Proposed Answer: C. 1) Both 2) lit

Explanation:

Candidate must understand the reason the reactor trip breakers would/would not be open and why SI would or would not actuate for a loss of 2 instrument buses in order to answer the question successfully.

A. Incorrect. Second part is correct. Loss of PY-14 causes a loss of one train of SI, however, SI actuates and both RTBs open.
B. Incorrect. Both RTBs open. One train of SI actuates and the PK will be lit due to the coincidence met for actuation due to the loss of two trains of bistables.
C. Correct. Both RTBs UV coils will de-energize and cause a reactor trip due to 2 trains of trip bistables tripping on the loss of power to the PY's. While one train of slave relays for SI is de-energized, SI actuates for the second train and PK08-21 will be lit.
D. Incorrect. First part correct. Second part is plausible due to loss of one train of slave relays and it could be thought that the second train is powered from PY-13.

Technical References: OIM B-6-1b, OP AP-4

References to be provided to applicants during exam: None

Learning Objective: Explain the consequences of loss of vital instrument bus. (4274)

Question Source:		
	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 3.2

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
APE 015 AA1.03 - Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Reactor trip alarms, switches, and indicators		**Group #**	1
		K/A #	APE 015 AA1.03
		Rating	3.7

Question 52

Unit 1 is at 40% power. Main Turbine load is 410 MWe.

RCP 1-1 trips.

1) As a result of the RCP 1-1 trip, PK04-14 REACTOR TRIP ACTUATED will __________ lit.

2) Indicated RCS flow in RCS Loop 1-1 will stabilize at approximately ____%.

A. 1) NOT be 2) 0

B. 1) NOT be 2) 30

C. 1) be 2) 0

D. 1) be 2) 30

Proposed Answer: D. 1) be 2) 30

Explanation

A. Incorrect. P-8, Loss of flow, will cause a reactor trip when REACTOR power is above 35%. If its thought its turbine power for P-8 (as it is with P-13 or C-5), then power would be below P-8 and would not cause a trip. Loop flow will lower but stabilize at approximately 30% due to reverse flow from the other loops.
B. Incorrect. First part incorrect, power is above P-8 and the RCP trip will cause a reactor trip. Second part is correct, flow lowers but because of reverse flow through the affected loop, flow will indicate approximately 30%.
C. Incorrect. First part is correct. Second part is incorrect. Flow will stabilize at approximately 30%.
D. Correct. Both parts correct. The reactor will trip and RCS flow, due to reverse flow, will indicate approximately 30%.

Technical References: LB-6A, OIM B 6-2, 6-3, 6-4a

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the Reactor Protection System. (37048)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.4

Examination Outline Cross-Reference	**Level**	RO
	Tier #	1
EPE 038 EA2.12 - Ability to determine or interpret the following as they apply to a SGTR: Status of MSIV activating system	**Group #**	1
	K/A #	EPE 038 EA2.12
	Rating	3.9

Question 53

GIVEN:

S/G	Narrow Range Level	Trend	Steam Generator Pressure	Trend
1-1	92%	Rising slowly	1040 psig	Stable
1-2	22%	Rising slowly	800 psig	Lowering slowly
1-3	22%	Rising slowly	800 psig	Lowering slowly
1-4	0%	Offscale	550 psig	Lowering rapidly

- The crew has performed EOP E-2, Faulted Steam Generator Isolation and has entered EOP E-3, Steam Generator Tube Rupture
- The operator has completed EOP E-3 Appendix FF, Isolate Flow from Ruptured Steam Generator, up through step 3, "CLOSE Ruptured S/Gs MSIV and MSIV Bypass Valves"

What is the current status of the Steam Generator MSIVs?

A. All MSIVs are closed.

B. Only Steam Generator 1-1 MSIV is closed.

C. Only Steam Generator 1-4 MSIV is closed.

D. Only Steam Generator 1-1 and 1-4 MSIVs are closed.

Proposed Answer: A. All MSIVs are closed.

Explanation:

A. Correct. MSI occurred due to Steam Generator 1-4 pressure. All MSIVs closed.
B. Incorrect. All MSIVs are closed. Plausible due to P-14, at 90%, causes auto actions to occur, however, closing MSIVs is not one of the auto actions.
C. Incorrect. Plausible if its thought that the low pressure in Steam Generator 1-4 closed only that MSIV.
D. Incorrect. All MSIVs are closed. Plausible if its thought P-14 closed the 1-1 Steam Generator MSIV and low pressure closed 1-4 Steam Generator MSIV and the pressures in 1-2 and 1-3 are the same (due to supplying TDAFW pump).

Technical References: OIM B-6-10, B-6-2, C-2-1, E-3 appendix FF

References to be provided to applicants during exam: None

Learning Objective: Analyze automatic features and interlocks associated with the Main Steam System. (7340)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 3.1

Examination Outline Cross-Reference	**Level**	RO
	Tier #	1
APE 056 G2.4.46 – Loss of Offsite Power: Ability to verify that the alarms are consistent with the plant conditions.	**Group #**	1
	K/A #	APE 056 G2.4.46
	Rating	4.2

Question 54

A loss of all offsite power occurs.

Emergency Diesel Generator 1-1 automatically starts and then trips.

1) What trip(s) could have caused Emergency Diesel Generator 1-1 to trip?

   1. Overspeed
   2. Jacket Water High temperature

2) What alarm(s) will be received in the Control Room?

   3. PK16-13, DIESEL 11 ENGINE TRIP
   4. PK16-15, DSL GEN 11 SHUTDOWN RELAY TRIP

A. 1) 1 only
   2) 3 only

B. 1) 1 only
   2) 3 and 4

C. 1) 1 and 2
   2) 3 only

D. 1) 1 and 2
   2) 3 and 4

Proposed Answer: B.1) 1 only 2) 3 and 4

Explanation:

A. Incorrect. First part is correct, only overspeed will cause the trip. Jacket water high temp is only active when the diesel is started in LOCAL. Second part is incorrect because both PK16-13 and PK16-15 will be in alarm.
B. Correct. Only the overspeed trip could cause the trip. Additionally, both PK16-13 and PK16-15 will alarm.
C. Incorrect Both parts incorrect. The jacket water high temperature trip is not active for automatic starts and both PKs will be in alarm.
D. Incorrect. First part incorrect. Second part is correct.

Technical References: AR PK16-13, 16-15

References to be provided to applicants during exam: None

Learning Objective: Describe controls, indications, and alarms associated with the Diesel Generator System. (37724)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.7

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
EPE 011 EK1.01 - Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA: Natural circulation and cooling, including reflux boiling		**Group #**	1
		K/A #	EPE 011 EK1.01
		Rating	4.1

Question 55

GIVEN:

- A large break LOCA has occurred
- RCS pressure is 25 psig
- Reactor Trip and Safety Injection have actuated
- All Centrifugal Charging Pumps have failed
- Safety Injection Pumps and Residual Heat Removal Pumps are running

How significant is reflux cooling in removing heat from the core for the current plant conditions?

A. Reflux cooling is minimal. Core heat is removed by Safety Injection water flowing out the break to the containment sump and eventually recirculated.

B. Reflux cooling is minimal. Safety injection pumps will provide enough volume to maintain the core covered, and heat will be removed using natural circulation and steam dumps.

C. Reflux cooling is significant. Without high pressure injection via the charging pumps, steam generator U-tubes will eventually void, stopping Natural Circulation.

D. Reflux cooling is significant. Coolant is boiled in the core, with the steam condensing in the Steam Generator tubes, flowing back to the core through the Hot Legs.

Proposed Answer: A. Reflux cooling is minimal. Core heat is removed by Safety Injection water flowing out the break to the containment sump and eventually recirculated

Explanation:

A. Correct. On a LBLOCA, the RCS rapidly depressurizes to containment pressure. The large injection and leak flow rates remove core decay heat. SG pressures are higher than RCS pressure. Reflux cooling is insignificant.
B. Incorrect. On a LBLOCA the RCS blows down and the SG tubes are voided. Natural circulation is not available. Plausible because reflux cooling is minimal, and ECCS pumps and accumulators will eventually refill and reflood the core.
C. Incorrect. Reflux cooling is insignificant on a LBLOCA. Plausible because on a Small Break LOCA without high pressure injection, natural circulation will be interrupted when SG tubes void, making reflux more important.
D. Incorrect. Reflux cooling is insignificant on a LBLOCA. Plausible because the description of reflux cooling is correct.

Technical References: LMCD-FRC

References to be provided to applicants during exam: None

Learning Objective: Explain how core cooling is provided during a loss of reactor coolant including the role of the following: …(d) Reflux Cooling…” (41698)
2. “Differentiate between large and small break LOCAs based on their characteristic stages.”

(41699)

Question Source:	Bank #41 L121 08/2014	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 08/2014	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Difficulty: 2.2

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
APE 065 AA1.03 - Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: Restoration of systems served by instrument air when pressure is regained		**Group #**	1
		K/A #	APE 065 AA1.03
		Rating	2.9

Question 56

Unit 1 is tripped due to a loss of instrument air.

Nitrogen is not available to a 10% steam dump valve.

1) Without instrument air or nitrogen, the 10% steam dump valve fails:

2) The operator can regain control of the valve by:

A. 1) open
   2) using the Cut-in toggle switch and the OPEN/CLOSE switch on VB3

B. 1) open
   2) placing the AUTO/MANUAL controller in MANUAL and using the increase/decrease pushbuttons on VB3

C. 1) closed
   2) using the Cut-in toggle switch and the OPEN/CLOSE switch on VB3

D. 1) closed
   2) placing the AUTO/MANUAL controller in MANUAL and using the increase/decrease pushbuttons on VB3

Proposed Answer: C. 1) closed
2) using the Cut-in toggle switch and the OPEN/CLOSE switch on VB3

Explanation:

A. Incorrect. First part incorrect, the valves fail closed. Second part correct.
B. Incorrect. The valves fail closed, not open. Operation must use the toggle switch, not the controller.
C. Correct. The loss of instrument air and nitrogen causes the valve to fail closed. Restoration of the 10% steam dumps is by cutting in backup air using the Cut In toggle switch. Backup air is supplied thru the cut-in switch and then controlled using the open/close switch on VB3. The controller is no longer part of the circuit.
D. Incorrect. First part is correct, the valve fails closed, but air to operate the valve is not thru the controller (controller only available with instrument air or nitrogen).

Technical References: LC-2B

References to be provided to applicants during exam: None

Learning Objective: Describe controls, indications, and alarms associated with the Steam DumpSystem. (37810)

Question Source:	Bank #54 L162 02/2018	X
(note changes; attach parent)	Modified Bank #	
	New	

	Past NRC Exam DCPP NRC 02/2018	Yes
Question History:	Last Two NRC Exams	Yes
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.3

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
APE 059 G2.4.45 - Accidental Liquid Radwaste Release: Ability to prioritize and interpret the significance of each annunciator or alarm.		**Group #**	2
		K/A #	APE 059 G2.4.45
		Rating	4.1

Question 57

Unit 1 is at 100% power.

PK11-17, SG BLOW DOWN HI RAD, input 508, Steam Gen Blowdown Sample Hdr Hi Rad, alarms.

Which of the following should have occurred?

NOTE:
- FCV-498, Disch Tunnel valve
- FCV-499, Equip Drn Rcvr valve

A. Both FCV-498 and FCV-499 closed

B. Both FCV-498 and FCV-499 opened

C. FCV-498 closed and FCV-499 opened

D. FCV-498 opened and FCV-499 closed

Proposed Answer: C. FCV-498 closed and FCV-499 opened

Explanation:
A. Incorrect. Plausible that both valves close to isolate the discharge.
B. Incorrect. FCV-499 opens.
C. Correct. Either RE-19 or 23 will isolate the system by closing the outside containment, sample valves and terminate blowdown effluent flow to the discharge tunnel by aligning to the Equipment Drain Receiver.
D. Incorrect. This is backwards of actual actions.

Technical References: LD-2, AR PK11-17

References to be provided to applicants during exam: None

Learning Objective: 8724 Analyze automatic features and interlocks associated with the SGBD system

Question Source:	Bank #59 DCPP L161 10/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 10/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.7	

Difficulty: 2.0

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
E16 EK1.3 - Knowledge of the operational implications of the following concepts as they apply to the (High Containment Radiation) Annunciators and conditions indicating signals, and remedial actions associated with the (High Containment Radiation)		**Group #**	2
		K/A #	E16 EK1.3
		Rating	3.0

Question 58

The crew is checking Pressurizer level in E-1, Loss of Reactor or Secondary Coolant, at step 7, "CHECK If ECCS Flow Should Be Reduced." PK11-19, CONTMT RADIATION, is LIT.

1. According to EOP F-0, Critical Safety Function Status Trees, the crew will use adverse containment values for Pressurizer level if the radiation rate setpoint of ____________ is reached.
2. If in adverse containment, the Pressurizer level required to reduce ECCS flow will be ___________ than the non-adverse setpoint value.

A. 1) 10^5 R/HR
   2) lower

B. 1) 10^5 R/HR
   2) higher

C. 1) 10^6 R/HR
   2) lower

D. 1) 10^6 R/HR
   2) higher

Proposed Answer: B. 1) 10^5 R/HR 2) higher

Explanation:

A. Incorrect. If containment radiation rate is greater than 10^5 R/HR, the operator is instructed to use adverse containment setpoints. However, the required Pressurizer level for reducing ECCS flow is higher (40%) than the normal value (12%).
B. Correct. Adverse containment setpoint is 10^5 R/hr and the level required is 40% vs 12%
C. Incorrect. Setpoint is 10^5 r/hr. 10^6 R is the integrated dose. Level required is higher, not lower.
D. Incorrect. Setpoint is incorrect. Required level is correct.

Technical References: EOP E-1, EOP F-0

References to be provided to applicants during exam: None

Learning Objective: Explain the use of adverse containment parameters during EOP usage

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X

	Comprehensive/Analysis
10CFR Part 55 Content:	55.41.10
Difficulty: 2.2	

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
APE 033 AK3.01 - Knowledge of the reasons for the following responses as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Termination of startup following loss of intermediate range instrumentation-		**Group #**	2
		K/A #	APE 033 AK3.01
		Rating	3.2

Question 59

GIVEN:
- Reactor startup in progress
- Reactor is critical in the intermediate range at $1x10^{-5}$ amps.

Intermediate Range channel N36 loses control power and fails low.

What is the effect on the reactor startup?

A. Delayed, the reactor remains critical, however, both Source Ranges are energized.

B. Delayed, the reactor remains critical, however, one Intermediate Range High Flux bistable has tripped.

C. Terminated, because the reactor has tripped because one Intermediate Range High Flux bistable has tripped.

D. Terminated, because the reactor has tripped due to the P-6 Permissive clearing and energizing the Source Range channels above the Source Range High Flux trip setpoint.

Proposed Answer: C.Terminated because the reactor has tripped due to the Intermediate Range High Flux trip bistable de-energized.

Explanation:

A. Incorrect. The reactor has tripped. P-6 will not have actuated and energized the source ranges.
B. Incorrect. The reactor will be tripped. One high flux bistable will be tripped, but the coincidence is 1/2. Trip coincidence could be confused with P-6 energizing the Source Ranges coincidence 2/2.
C. Correct. when the IR loses control power, the High Flux trip (1 of 2) will cause the reactor to trip.
D. Incorrect. P-6 automatically energizes the Source Ranges on a downpower, but it is a 2/2 coincidence (1 of 2 raising power), one channel low will not cause P-6 to energize SR.

Technical References: OIM B-4-2

References to be provided to applicants during exam: None

Learning Objective: Explain the effect of Excore NIS channel failures, including:
• Individual Channel Failures
• Loss of power supplies/fuses

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #59 DCPP NRC 04/2007	X
	New	
	Past NRC Exam DCPP NRC 04/2007	Yes
Question History:	Last Two NRC Exams	No

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.2	

Difficulty: 2.5

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
APE 061 AA2.01 - Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: - ARM panel displays		**Group #**	2
		K/A #	APE 061 AA2.01
		Rating	3.5

Question 60

GIVEN:

- Radiation levels begin rising in the Fuel Handling Building
- At 1300 RM-58, Fuel Handling Building (FHB) Radiation Monitor, reaches the Alert setpoint (Amber light lit)
- At 1305, RM-58, Fuel Handling Building (FHB) Radiation Monitor, reaches the High setpoint (Red light lit)

1) What time did the FHB automatic ventilation changes associated with RM-58 occur?

2) An automatic action that occurs is the:

A. 1) 1300
   2) supply fan(s) stop

B. 1) 1300
   2) exhaust air is routed through a charcoal filter

C. 1) 1305
   2) supply fan(s) stop

D. 1) 1305
   2) exhaust air is routed through a charcoal filter

Proposed Answer: D. 1) 1305 2) exhaust air is routed through a charcoal filter

Explanation:

A. Incorrect. Both parts incorrect. Auto action occurs at the high (Trip 2) setpoint and ventilation realigns such that exhaust is routed through a charcoal filter by starting exhaust fan E-5 or E-6. A supply fan continues to run.
B. Incorrect. First part incorrect. Second part is correct.
C. Incorrect. First part is correct. Ventilation switches to Iodine Removal at the Trip 2 setpoint. Second part is incorrect. Plausible because it may be thought that supply fans should be stopped to force a negative pressure to prevent radioactivity from the room from exiting anywhere except out the ventilation system past effluent radiation monitors. However, fan capacities ensure a negative pressure with one supply and one exhaust fan running.
D. Correct. Exhaust fan E-5 or E-6 start and exhaust is routed through a charcoal filter to limit the radiation released.

Technical References: LH-7, LG-4A

References to be provided to applicants during exam: None

Learning Objective: Discuss abnormal conditions associated with the Fuel Handling Building Ventilation System. (40721)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X

	Past NRC Exam	
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.11	

Difficulty: 2.1

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
E13 EA1.2 - Ability to operate and / or monitor the following as they apply to the (Steam Generator Overpressure) Operating behavior characteristics of the facility		**Group #**	2
		K/A #	E13 EA1.2
		Rating	3.0

Question 61

1) If steam generator pressure rises to 1120 psig, the Critical Safety Function Status tree terminus for Heat Sink will have turned from Green to ______________.

2) At steam generator pressure of 1120 psig, how many of the steam generator safeties should be open?

A. 1) Red
   2) one

B. 1) Red
   2) five

C. 1) Yellow
   2) one

D. 1) Yellow
   2) five

Proposed Answer: D. 1) Yellow 2) five

Explanation:

A. Incorrect. The heat sink CSF is either Red or Yellow. All the safeties should be open. Plausible if setpoints are not known. One safety is a plausible cause of a challenge to the safety function. Red is plausible as other CSFs have more than one red path entry, such as RCS Integrity and Containment Integrity.
B. Incorrect. First part is incorrect. At 1120 psig, all safeties should be open.
C. Incorrect. First part correct. The terminus will be yellow. Second part incorrect. All safeties should be open.
D. Correct. The Heat Sink terminus is Red, Yellow, or Green. If pressure rises to 1115 psig, the terminus will be yellow. Safety setpoints are 1065 psig, 1078 psig, 1090 psig, 1103 psig and 1115. All should be open.

Technical References: F-0

References to be provided to applicants during exam: None

Learning Objective: Describe Main Steam System components.

- Steam Line Safety Valves

Question Source:		Bank #	
	(note changes; attach parent)	Modified Bank #64 DCPP L161 10/2016	X
		New	
		Past NRC Exam DCPP 10/2016	Yes
Question History:		Last Two NRC Exams	No
Question Cognitive Level:		Memory/Fundamental	
		Comprehensive/Analysis	X

10CFR Part 55 Content: 55.41.8
Difficulty: 3.0

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
APE 001 AA2.05 - Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Uncontrolled rod withdrawal, from available indications		**Group #**	2
		K/A #	APE 001 AA2.05
		Rating	4.4

Question 62

Unit 1 is at 65% power. Control systems are in AUTO.

Rods begin to step out at 8 steps per minute.

The following alarms are received:
- PK04-03, TAVG DEVIATION FROM REF
- PK05-16, PZR PRESSURE HI/LO, due to input 363, "Pzr Press Hi From REF"

The operator reports Tref is unchanged.

Which of the following actions should be taken by the operator?

A. Trip the reactor due to dropped rods.

B. Trip the reactor due to a vapor space leak.

C. Place rods in Manual due to a malfunction of rod control.

D. Place rods in Manual due to a power range instrument failure.

Proposed Answer: C. Place rods in Manual due to a malfunction of rod control.

Explanation:
A. Incorrect. Dropped rods would not cause high pressure. Plausible because the reactor is tripped if 2 or more rods drop.
B. Incorrect. No indication of trip required, a vapor space break causes pressurizer level to rise, but does not impact temperature.
C. Correct. Indication of outward rod motion. Action is to place rods in Manual.
D. Incorrect. No power range instrument failure would cause high pressure, high level and increasing temperature and outward rod motion.

Technical References: OP AP-12A, Continuous Withdrawal or Insertion of a Control Rod Bank.

References to be provided to applicants during exam: None

Learning Objective: Given an abnormal condition, summarize the major actions of OP AP-12 to mitigate an event in progress. (3477M)

Question Source:	Bank #57 L051 04/2007	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 04/2007	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	

	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	
Difficulty: 2.4		

Examination Outline Cross-Reference	Level	RO
	Tier #	1
APE 037 G2.4.8 Steam Generator Tube Leak- Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	Group #	2
	K/A #	APE 037 G2.4.8
	Rating	3.8

Question 63

The crew has started a plant shutdown in accordance with OP AP-3, Steam Generator Tube Failure.

According to OP AP-3, the procedure must be completed unless:

1) the reactor trips
2) superseded by OP AP-1, Excessive Reactor Coolant System Leakage
3) superseded by EOP E-3, Steam Generator Tube Rupture

A. 1 only

B. 3 only

C. 1 and 2 only

D. 2 and 3 only

Proposed Answer: B. 3 only

Explanation:

A. Incorrect. The procedure is completed unless superseded by EOP E-3. Plausible because reactor trip requires entry into the EOP network and other AOPs, such as AP-1 do not have the requirement to finish if the reactor trips.
B. Correct. The note states: “Once it has been determined that a leaking S/G exists that would require shutdown per this procedure, this procedure must be completed unless superseded by EOP E-3. If the EOPs are entered for any reason during performance of this procedure, the actions required by this procedure must be completed upon exiting the Emergency Procedures. Procedures may be performed in parallel if resources allow.
C. Incorrect. Either is plausible as procedures that could cause stopping AP-3.
D. Incorrect. Plausible as RCS leakage and a tube rupture in the EOP network would require leaving E-3 and going to ECA-3.1 Could be thought that the complication of RCS leakage would terminate the necessity to complete OP AP-3 actions.

Technical References: OP AP-1, OP AP-3

References to be provided to applicants during exam: None

Learning Objective: Given an abnormal condition, summarize the major actions of OP AP-3 to mitigate an event in progress. (3477C)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X

Comprehensive/Analysis

10CFR Part 55 Content: 55.41.10

Difficulty: 2.0

Examination Outline Cross-Reference	**Level**	RO
	Tier #	1
EPE 074 EK1.03 - Knowledge of the operational implications of the following concepts as they apply to the Inadequate Core Cooling: Processes for removing decay heat from the core	**Group #**	2
	K/A #	EPE 074 EK1.03
	Rating	4.5

Question 64

According to the background document for EOP FR-C.1, Response to Inadequate Core Cooling, what is the most effective method to recover the core and restore core cooling?

A. Initiation of RCS feed and bleed.

B. Starting RCPs in available RCS loops.

C. Establishing high head ECCS injection.

D. Rapidly depressurizing steam generators.

Proposed Answer: C. Establishing high head ECCS injection

Explanation:

A. Incorrect. This is the method used in EOP FR-H.1 – as a last resort.
B. Incorrect. Starting RCPs is a temporary method to core until some form of flow can be established.
C. Correct. IAW the background document: *Reinitiation of high pressure safety injection is the most effective method to recover the core and restore adequate core cooling. If some form of high pressure injection cannot be established or is ineffective in restoring adequate core cooling, then the operator must take actions to reduce the RCS pressure in order for the SI accumulators and low-head SI pumps to inject.* Analyses have shown that a rapid secondary depressurization is the most effective means for achieving this. If secondary depressurization is not possible, or primary-to-secondary heat transfer is significantly degraded, and at least one idle SG is available, then the operator must start the RCP(s) associated with the available idle SG(s). The RCPs will provide forced two phase flow through the core and temporarily improve core cooling until some form of make-up flow to the RCS can be established.
D. Incorrect. Depressurization of the RCS to establish low head injection flow (and inject accumulators is the preferred method of heat removal if high head injection cannot be established).

Technical References: EOP FR-C.1 background

References to be provided to applicants during exam: None

Learning Objective: Prioritize the operator-initiated recovery techniques that would mitigate the consequences of a loss of core cooling. (11311)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.5	

Difficulty: 2.3

Examination Outline Cross-Reference		**Level**	RO
		Tier #	1
APE 060 AK2.02 - Knowledge of the interrelations between the Accidental Gaseous Radwaste Release and the following: Auxiliary building ventilation system		**Group #**	2
		K/A #	APE 060 AK2.02
		Rating	2.7

Question 65

PK11-25, PLANT VENT RADIATION and PK11-21, HIGH RADIATION alarm. The source of the radiation is unknown.

The SFM directs operators to enter OP AP-14, Tank Ruptures.

In accordance with OP AP-14, how will the crew adjust Auxiliary Building ventilation?

A. Select "Building and Safeguards" mode.

B. Select "Safeguards Only" with "S" signal.

C. Select "Safeguards Only" without "S" signal.

D. Stop all running all supply and exhaust fans.

Proposed Answer: B. Select "Safeguards Only" with "S" signal.

Explanation:

A. Incorrect. Building and Safeguards is a mode selected when an ECCS component is running. Plausible if the modes are not understood.
B. Correct. This aligns ventilation through the charcoal filter for removal of contamination.
C. Incorrect. This does not align the ventilation through the charcoal filter but plausible if modes not understood.
D. Incorrect. Stopping all could be thought as a method to stop the spread of contamination.

Technical References: LH-1, OIM H-2-4, H-2-5

References to be provided to applicants during exam: None

Learning Objective: Describe the operation of the Auxiliary Building Ventilation System. (5512)

Question Source:		Bank #	
X	(note changes; attach parent)	Modified Bank #	
		New	X
		Past NRC Exam	No
Question History:		Last Two NRC Exams	No
Question Cognitive Level:		Memory/Fundamental	
		Comprehensive/Analysis	X
10CFR Part 55 Content:		55.41.11	

Difficulty: 2.0

Examination Outline Cross-Reference	**Level**	RO
	Tier #	3
G2.1.44 Knowledge of RO duties in the control room during fuel handling, such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.	**Group #**	1
	K/A #	G2.1.44
	Rating	3.9

Question 66

While taking logs during core loading the operator notes the following trends.

Time	RCS Temp	Boron	N31	N32
1200	62°F	2570 PPM	13 CPS	15 CPS
1300	76°F	2510 PPM	32 CPS	28 CPS

The recommendation to the Shift Foreman is that core loading should be suspended due to changes in ___________________________.

A. RCS temperature

B. boron concentration

C. counts on one Source Range channel

D. counts on both Source Range channels

Proposed Answer: B. boron concentration

Explanation:
step 5.3.2 - The loading procedure will be suspended, pending evaluation by the Refueling SRO and reactor engineer under the following circumstances:
a. If there occurs on any one responding nuclear channel an unexpected increase in count rate by a factor of three (3).
b. An unexpected increase in count rate by a factor of two on all responding channels.
c. An unexpected change in Reactor Coolant System temperature of greater than 20°F.
d. If the measured boron concentration indicates a change of greater than ± 50 ppm from the nominal value at the start of core loading.
A. Incorrect. Temperature change is less than 20°F (14°F)
B. Correct. Boron change is greater than 50 ppm (60 ppm)
C. Incorrect. Source range counts on one channel is less than 3 (approximately 2.5 – 13 to 32)
D. Incorrect. Counts did not double on both SR channels (not doubled on N32)

Technical References: OP B-8DS2

References to be provided to applicants during exam: None

Learning Objective: 36965 - Discuss significant precautions and limitations associated with the Fuel Handling system

Question Source:	Bank #60 L061C 02/2009	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 02/2009	Yes

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.6	

Difficulty: 2.6

Examination Outline Cross-Reference		**Level**	RO
		Tier #	3
G2.2.40 Ability to apply Technical Specifications for a system.		**Group #**	2
		K/A #	G2.2.40
		Rating	3.4

Question 67

1) LCO 3.2.4, QUADRANT POWER TILT RATIO (QPTR), states QPTR shall be ≤ ______ when power THERMAL POWER is > 50% RTP.

2) According to Technical Specifications section 1.1, Definitions, the definition of QPTR is which of the following?

   a. the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

   b. the difference in normalized flux signals between the top and bottom halves of an excore neutron detector.

A. 1) 1.00
   2) a

B. 1) 1.00
   2) b

C. 1) 1.02
   2) a

D. 1) 1.02
   2) b

Proposed Answer: C. 1) 1.02 2) a.

Explanation:

A. Incorrect. LCO 3.2.4 states QPTR shall be less than or equal to 1.02. Plausible the limit is less than 1 as the action is to reduce power 3% for every 1% greater than *1.00* Second part correct.
B. Incorrect. First part incorrect. Second part incorrect, this is the definition for AFD.
C. Correct. LCO limit is 1.02. Second part is the definition for QPTR.
D. Incorrect. First part correct. Second part incorrect – this is the definition for AFD

Technical References: LCO 3.2.4, Section 1.1, definitions

References to be provided to applicants during exam: None

Learning Objective: 9697 - Apply Technical Specification LCOs

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Difficulty: 2.7

Examination Outline Cross-Reference		**Level**	RO
		Tier #	3
G2.2.2 Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.		**Group #**	2
		K/A #	G2.2.2
		Rating	4.6

Question 68

Which of the following describes a normally performed operator action as a plant startup from MODE 5 to MODE 2 is performed?

A. When PK08-01, INTMED RNG PERMISSIVE P-6, goes ON, the operator takes the Source Range Trip Reset/Block switches to “BLOCK”.

B. When PK08-05, PWR RNG AT POWER PER P-10, goes ON, the operator takes the Intermediate Range Rod Stop and Trip Block switches to “BLOCK”.

C. When RCS pressure exceeds P-11 and PK08-06, PZR SI PERMISSIVE, P-11, goes OFF, the operator takes the Pressurizer SI RESET/BLOCK switches to “RESET”.

D. When RCS temperature exceeds 283°F and RHR has been removed from service, the operator takes the Low Setpoint Protection Cutout switches to CUTIN for the Pressurizer PORVs.

Proposed Answer: A. When PK08-01, INTMED RNG PERMISSIVE P-6, goes ON, the operator takes the Source Range Trip Reset/Block switches to “BLOCK”.

Explanation:

A. Correct. At P-6, the operator goes to BLOCK to de-energize the Source Range instruments - OP L-2, attachment 3, 1.c
B. Incorrect. This action is not taken at this time. This would block the IR High Flux trip. This is done at P-10, which is at 10% power (in MODE 1).
C. Incorrect. This is normally not done. SI should auto unblock as pressure is raised. When performing a cooldown, the SI signals are blocked (action required), however, no action should be required if there is not a failure of the circuit. Setup notes it’s a normal action that is taken – OP L-1 step 6.2.3.h.3
D. Incorrect. To remove LTOP from service, the switches are placed in CUTOUT (placed in CUTIN to place in service when performing cooldown) – OP L-1, step 6.1.3.m.

Technical References: OP L-1, OP L-2, L-3, Callaway OE 2019 Trip during startup

References to be provided to applicants during exam: None

Learning Objective: Discuss operator behaviors and practices related to the operator fundamental of closely monitoring plant indications and conditions. (56218)

Question Source:	Bank #70 DCPP NRC L121 08/2014	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 08/2014	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.6	

Difficulty: 2.5

Examination Outline Cross-Reference	Level	RO
	Tier #	3
G2.4.17 Knowledge of EOP terms and definitions.	Group #	4
	K/A #	G2.4.17
	Rating	3.9

Question 69

In accordance with AD1.DC12, Writer's Manual - Emergency and Off-Normal Operating Procedures:

1) a block around a step number identifies the step as a(n) ______________ action.

2) the term VERIFY has the same meaning as ______________.

A. 1) immediate
   2) CHECK

B. 1) immediate
   2) ENSURE

C. 1) continuous
   2) CHECK

D. 1) continuous
   2) ENSURE

Proposed Answer: B.1) immediate 2) ENSURE

Explanation:

A. Incorrect. First part is correct. An immediate action is identified with a block around the step number. Second part incorrect. CHECK means: To note a condition and compare with some procedure requirement. Verify has the same meaning as Ensure.
B. Correct. An immediate action is identified with a block around the step number. Ensure means to make certain a certain characteristic or condition exists by either confirming the condition, or taking the necessary actions to establish the condition. Verify has the same meaning.
C. Incorrect. Both parts are incorrect. A block around the entire step is used to identify a "continuous action". Verify is the same as Ensure.
D. Incorrect. First part is incorrect, a block around an entire step is used for "continuous actions".Second part is correct.

Technical References: AD1.DC12 and attachment 1

References to be provided to applicants during exam: None

Learning Objective: Explain definition of terms used in the EOPs. (5428)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X

	Comprehensive/Analysis
10CFR Part 55 Content:	55.41.10
Difficulty: 2.0	

Examination Outline Cross-Reference		**Level**	RO
		Tier #	3
G2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions		**Group #**	2
		K/A #	G2.2.44
		Rating	4.2

Question 70

The crew is reviewing the foldout page as part of a procedure transition brief.

In accordance with OP1.DC10, Conduct of Operations, as a minimum, what amount of communication is required by an operator who is assigned a foldout page item to monitor?

A. Repeat back of the high level action

B. Simple acknowledgement of the assignment

C. A brief summary of the action and the parameters to monitor

D. Repeat back of the high level action and the specific parameters and parameters to monitor

Proposed Answer: A. Repeat back of the high level action

Explanation:
A. Correct. Per OP1.DC10,
a) Specific assignments should be made to Control Room operators by assigning the foldout page number and the *operator repeating back the high-level action*.
b) Specific parameters and values are not required to be repeated back.
c) A copy of the foldout page should be given to any operator with an assignment.
B. Incorrect. The high level action is repeated back.
C. Incorrect. Only the high level action needs to be repeated back.
D. Incorrrect. Only the high level action needs to be repeated back.

Technical References: OP1.DC10

References to be provided to applicants during exam: None

Learning Objective: 41675 - Describe the expectations for performing briefs and updates, including the following: Procedure transition briefs, Pre-job briefs, Diagnostic briefs

Question Source:	Bank #67 DCPP L091 07/2011	Yes
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 07/2011	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Difficulty: 2.0

Examination Outline Cross-Reference		**Level**	RO
		Tier #	3
G2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.		**Group #**	3
		K/A #	G2.3.12
		Rating	3.2

Question 71

Unit 1 is at 100% power.

An operator is preparing to enter Unit 1 containment for a non-emergency entry.

In accordance with RCP D-230, Radiological Control for Containment Entry, as part of exposure control for the operator, the __________________________ shall maintain possession of the MIDS keys during the containment entry?

A. Operator making the entry

B. Unit 1 Shift Foreman (or designee)

C. Work Control Shift Foreman (or designee)

D. Radiation Protection Foreman (or designee)

Proposed Answer: D. Radiation Protection Foreman (or designee)

Explanation:

A. Incorrect. While it may seem that having the operator control the key would be a positive control, the keys shall be in the possession of the RP Foreman (or designee)
B. Incorrect. The SFM authorizes entry and controls keys, but not the MIDS keys.
C. Incorrect. The WCSFM is responsible for authorizing work packages, but does not control the MIDS keys
D. Correct. The RP foreman shall be in possession of the keys (or designee).

Technical References: RCP D-230

References to be provided to applicants during exam: None

Learning Objective:

Question Source:	Bank #73 DCPP L141 04/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 04/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.12	

Difficulty: 2.0

Examination Outline Cross-Reference		**Level**	RO
		Tier #	3
G2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.		**Group #**	4
		K/A #	G2.4.35
		Rating	3.8

Question 72

The crew is performing EOP ECA-0.0, Loss of Vital AC Power.

An operator has been dispatched to perform Appendix DC, Shed Non-Essential DC Loads.

In accordance with Appendix DC,

1) the operator will stop the Main Turbine DC Bearing Oil Pump when __________________________.

2) the operator will stop the Air Side Seal Oil Backup Pump when __________________________.

A. 1) Main Turbine speed is zero
   2) Main Turbine speed is zero

B. 1) Main Turbine speed is zero
   2) the emergency Purge of the Main Generator is complete

C. 1) the emergency Purge of the Main Generator is complete
   2) Main Turbine speed is zero

D. 1) the emergency Purge of the Main Generator is complete
   2) the emergency Purge of the Main Generator is complete

Proposed Answer: B. 1) Main Turbine speed is zero
2) the emergency Purge of the Main Generator is complete

Explanation:

A. Incorrect. First part is correct. Second part is incorrect. Plausible if its known both pumps are in service but its not known the purge must be done.
B. Correct. Both are in service while the turbine coasts down. At zero speed, the operator is instructed to shutdown the DC Bearing Oil Pump (open breaker 72-1008). When the purge is compete, the Air Side Seal Oil pump is shutdown (open breaker 72-1006).
C. Incorrect. Both parts are incorrect. This is the reverse of the correct answer
D. Incorrect. Second part is correct. First part incorrect..

Technical References: EOP ECA-0.0, appendix DC

References to be provided to applicants during exam: None

Learning Objective: Explain the basis for securing DC loads on loss of vital AC bus. (7118)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.41.10	

Difficulty: 2.0

Examination Outline Cross-Reference		**Level**	RO
		Tier #	3
G2.1.26 Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).		**Group #**	1
		K/A #	G2.1.26
		Rating	3.4

Question 73

An operator calls and reports the spill of hydrazine in the vicinity of the Auxiliary Feedwater pump chemical addition skid.

In accordance with CP M-9A, Hazardous Materials Incident Initial Emergency Response/Mitigation Procedure, assistance should be immediately requested from:

A. DCPP Safety

B. Radiation Protection

C. DCPP Fire Department

D. Chemistry and Environmental Operations

Proposed Answer: C. DCPP Fire Department

Explanation:

A. Incorrect. Plausible that Safety would be the department to deal with chemical spills.
B. Incorrect Plausible because the spill in the Auxiliary Building (inside the RCA).
C. Correct. The procedure states to call the DCPP Hazardous Materials Emergency Response Team – which if the DCPP Fire Department.
D. Incorrect. Because it's a chemical spill, plausible Chemistry is contacted..

Technical References: CP M-9A

References to be provided to applicants during exam: None

Learning Objective: 39579 Discuss the appropriate response to various chemical spills

Question Source:	Bank #66 DCPP L091C 02/2009	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 02/2009	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Difficulty: 2.0

Examination Outline Cross-Reference		**Level**	RO
		Tier #	3
G2.1.37 - Knowledge of procedures, *guidelines*, or limitations associated with reactivity management.		**Group #**	1
		K/A #	G2.1.37
		Rating	4.3

Question 74

The operator at the controls notes reactor power is beginning to lower.

In accordance with OP1.DC10, Conduct of Operations, Attachment 11, Operations Diagnostic Model, the operator will check ______________________next.

A. RCS Tave

B. RCS subcooling

C. Pressurizer level

D. Pressurizer pressure

Proposed Answer: A. RCS Tave

Explanation:

Questions tests the guidelines in OP1.DC10 for monitoring the reactivity control.
Monitoring Key Parameters:

• POWER - *Reactivity control is of paramount importance. Reactor power could be changed by actions taken in the RCS* - such as boration or dilution - or by a very limited number of primary side problems (rod misalignment, for example). Reactor power could also be changed by changes in secondary power or efficiency. In support of cause and effect, power can be thought of as reactor power, or secondary power - but in essence it is the comparison between power produced in the reactor, and power removed by the secondary. *Changes in either the primary or secondary always result in changes to Tave*. Unintentional power changes must be immediately addressed.

• TEMPERATURE - *Changes to Tave only occur because of changes to the thermodynamic balance of the plant (i.e., the power produced by the reactor is different than the power removed by the secondary plant).* Tave changes always result in changes to the density of the RCS fluid inventory. Significant changes in density (Larger Tave changes) always result in changes to PZR level. Unintentional Tave changes require an understanding of POWER before proper action can be taken to stabilize the plant.

By quickly reviewing reactor and secondary power indications, then RCS Tave, then PZR level, then PZR pressure, then primary and secondary flow rates, a mental model (or image) of the true plant condition can be determined. If a key parameter alarms, simply reviewing the key parameters can help determine if the cause of the alarm can be attributed to the alarming parameter, or if the parameter transient is merely an effect of a higher priority parameter transient.

• When a plant transient occurs, the key parameters are quickly and frequently checked to ensure that proper system response is occurring. Provided that all systems are responding appropriately, no operator action would be required to essentially stabilize the plant. Stabilizing the highest priority parameter that is not stable aids in stabilizing all the lesser priority parameters (i.e., until power is stabilized, Tave cannot be stabilized. Until Tave is stabilized, PZR level cannot be stabilized, etc.).

A. Correct the next parameter is RCS Tave.
B. Incorrect.. Plausible because Subcooling is a key parameter checked in EOPs for verification of sufficient core cooling. However, it is not part of the PTLPF model
C. Incorrect level is checked after RCS Tave.
D. Incorrect. Pressurizer pressure is checked after Pressurizer level.

Technical References: OP1.DC10

References to be provided to applicants during exam: None

Learning Objective: Discuss the STAR-T diagnostic model. (56221)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #68 L161 10/2016	X
	New	
	Past NRC Exam DCPP 10/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.2	

Difficulty: 2.3

Examination Outline Cross-Reference		**Level**	RO
		Tier #	3
G2.3.11 Ability to control radiation releases.		**Group #**	3
		K/A #	G2.3.11
		Rating	3.8

Question 75

In accordance with OP G-1:II, Liquid Radwaste System - Discharge of Liquid Radwaste, what is REQUIRED prior to initiating a liquid radwaste discharge?

1. Sufficient dilution flowrate
2. Shift Foreman review and approval of the discharge permit
3. Shift Manager review and approval of the discharge permit

A. 2 only

B. 1 and 2

C. 3 only

D. 1 and 3

Proposed Answer: B.1 and 2

Explanation:

A. Incorrect because both a gas and liquid discharge require adequate dilution flow prior to initiating the discharge. Plausible because it may be thought that since review and approval is needed prior to initiating the release, and there must be an operable rad monitor (or comp actions in place), the review covers the need for dilution flow.
B. Correct. Approval by the SFM is required, additionally, there must be adequate flow rate (ASW or Circ water flow) for the discharge to occur.
C. Incorrect because review and approval is required, but not from the SM. Plausible because the SM is responsible for both units and radwaste release is a common function.
D. Incorrect because SM approval not required. Plausible because the SM is responsible for both units and a radwaste release is a common function.

Technical References: OP G-1:II

References to be provided to applicants during exam: None

Learning Objective: Discuss significant precautions and limitations associated with the Liquid Radwaste System. (8454)

Question Source:	Bank #71 DCPP L161	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 10/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.12	

Difficulty:2.0

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
004 A2.27 - Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations Improper RWST boron concentration	**Group #**	1
	K/A #	004 A2.27
	Rating	4.2

Question 76

Unit 1 is at 100% power.

Results of a sample of the RWST boron concentration reveals the boron concentration is higher than the limit of LCO 3.5.4, Refueling Water Storage Tank (RWST).

1) According to the bases for LCO 3.5.4, what is the potential impact of a boron concentration higher than the limit of LCO 3.5.4?
2) What action should be taken by the Shift Foreman?

NOTE:
- LCO 3.5.2, ECCS-Operating
- LCO 3.5.4, Refueling Water Storage Tank (RWST)

A. 1) excessive caustic stress corrosion of the RWST
   2) Enter only LCO 3.5.4

B. 1) excessive caustic stress corrosion of the RWST
   2) Enter LCO 3.5.2 and LCO 3.5.4

C. 1) excessive boric acid precipitation in the core following a large break LOCA
   2) Enter only LCO 3.5.4

D. 1) excessive boric acid precipitation in the core following a large break LOCA
   2) Enter LCO 3.5.2 and LCO 3.5.4

Proposed Answer: C. 1) excessive boric acid precipitation in the core following a large break LOCA
2) Enter only LCO 3.5.4

Explanation:

SRO knowledge of bases of LCO and applicability of LCO 3.0.2 to time of discovery.

A. Incorrect. First part is incorrect. Plausible as high boron can cause excessive caustic stress corrosion but of components in containment during a LOCA. Second part correct. LCO 3.5.2 does not need to be entered. Due to LCO 3.0.6.
B. Incorrect. Both parts are incorrect..
C. Correct. The bases for LCO 3.5.4 states During accident conditions, the RWST provides a source of borated water to the ECCS and CS System pumps. Improper boron concentrations could result in a reduction of SDM or *excessive boric acid precipitation* in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment. The upper limit on boron concentration of 2500 ppm is used to determine the maximum allowable time to initiate hot leg recirculation following a LOCA. The purpose of initiating hot leg recirculation is to avoid boron

precipitation in the core following the accident when the break is in the cold leg.

D. Incorrect. First part is correct. Second part incorrect. LCO 3.5.2 is not entered. Plausible because both trains of ECCS are supplied by the RWST and two inoperable trains of ECCS would require action

Technical References: B3.5.4, LCO 3.5.4, LCO 3.0.6

References to be provided to applicants during exam: none

Learning Objective: 9694E - Apply TS 3.5 Technical Specification bases

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	

Difficulty: 3.2

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
039 G2.1.7 – Main and Reheat Steam: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	**Group #**	1
	K/A #	039 G2.1.7
	Rating	4.7

Question 77

GIVEN:

- The crew is performing the diagnostic steps of EOP E-0, Reactor Trip or Safety Injection
- RCS temperature is currently 315°F and rising after an initial decrease to 230°F
- RCS pressure 1750 psig and rising
- Pressurizer level 15% and rising
- Steam Generator Narrow Range levels:
  - 1-1 is 22%, rising slowly
  - 1-2 is 0%, stable
  - 1-3 is 18%, stable
  - 1-4 is 23%, rising slowly
- Steam Generator pressures:
  - 1-1 is 820 psig, rising
  - 1-2 is 0 psig, stable
  - 1-3 is 900 psig, stable
  - 1-4 is 890 psig, rising

What procedure should the crew transition to next?

A. EOP E-1.1, Safety Injection Termination

B. EOP E-2, Faulted Steam Generator Isolation

C. EOP FR-H.5, Response to Steam Generator Low Level

D. EOP FR-P.1, Response to Imminent Pressurized Thermal Shock Condition

Proposed Answer: B.EOP E-2, Faulted Steam Generator Isolation

Explanation:

SRO must evaluate the current conditions and determine which procedure entry is appropriate. Based on conditions, a RED challenge to RCS Integrity had existed, but according to rules of EOP usage, does not need to be addressed. Also, SI termination criteria is met, however, a faulted S/G must be addressed prior to terminating SI>

A. Incorrect. Although conditions support SI termination, a transition to E-2 will be made based on Steam Generator 1-2 completely depressurized.
B. Correct. A transition from E-0 is required to perform the faulted steam generator isolation.
C. Incorrect. Conditions met for entry but H.5 is a YELLOW path and not addressed at this time.
D. Incorrect. The initial decrease was to a temperature that made the status tree for RCS integrity Red, however, the status is currently Green. The condition has cleared and does need to be addressed.

Technical References: F-0, E-0

References to be provided to applicants during exam: None

Learning Objective: 3552 - Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

Question Source:	Bank #85 DCPP L081 01/2010	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 01/2010	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3.0

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
061 A2.07 - Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Air or MOV failure	**Group #**	1
	K/A #	061 A2.07
	Rating	3.5

Question 78

GIVEN:

- Unit 1 trips from 100% power
- The crew is performing EOP E-0, Reactor Trip or Safety Injection
- 480VAC power is lost to AFW pump 1-1 LCVs
- None of the MDAFW pumps can be started
- Narrow range steam generator levels are 12% and lowering slowly in all Steam Generators
- Containment pressure is 0.8 psig and stable
- RCS temperature is 540°F and lowering slowly
- RCS pressure is 1900 psig and lowering slowly

Which of the following actions should be taken by the Shift Foreman as a transition is made from EOP E-0?

A. Go to EOP E-0.1, Reactor Trip Response, the turbine driven AFW pump is supplying all four steam generators.

B. Go to EOP E-0.1, Reactor Trip Response, and direct the operator to locally open the TDAFW level control valves, LCV-106, 107, 108 and 109 to establish AFW flow from the TDAFW pump.

C. Go to EOP FR-H.1, Response to Loss of Secondary Heat Sink, there is a RED path on Secondary Heat Sink because the TDAFW Steam Supply valve, FCV-95 did not open.

D. Go to EOP FR-H.1, Response to Loss of Secondary Heat Sink, there is a RED path on Secondary Heat Sink because the TDAFW level control valves, LCV-106, 107, 108 and 109 are closed.

Proposed Answer: A. Go to EOP E-0.1, Reactor Trip Response, the turbine driven AFW pump is supplying all four steam generators.

Explanation:

A. Correct. The TDAFW pump is running, it starts on 2/4 steam generators less than 15%.The TDAFW pump LCVs are powered from 480 VAC (Bus G), however, are left full open and will remain open. Although there is no valve indication (lights will be out), full flow from the pump will be supplying all four steam generators.
B. Incorrect. The valves will be fully open due to the loss of 480 VAC power and will have to locally throttled when heat sink is greater than 16%.
C. Incorrect. FCV-95 is DC powered. Loss of DC bus 1-2 would cause FCV-95 to lose power and Bus G would be de-energized, which causes the LCVs to lose power. However, the DC bus is energized, only the bus is de-energized.
D. Incorrect. The LCVs are powered from bus G but remain open despite the loss of power to

them, therefore, there is no challenge to Heat Sink CSF.

Technical References: OP AP-23

References to be provided to applicants during exam: None

Learning Objective: 3552 - Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

Question Source:	Bank #89 DCPP L091C 03/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 03/2012	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3.0

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
063 G2.2.25 – DC Electrical Distribution: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	**Group #**	1
	K/A #	063 G2.2.25
	Rating	4.2

Question 79

Unit 1 is at 100% power. One battery charger is declared inoperable.

LCO 3.8.4, DC Sources – Operating, Condition A is shown below:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One battery charger inoperable.	A.1 Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	2 hours
	AND	
	A.2 Verify battery float current ≤ 2 amps.	12 hours
	AND	
	A.3 Restore battery charger to OPERABLE status.	14 days

According to the bases for REQUIRED ACTION A.2, what is the reason for verifying battery float current ≤ 2 amps?

A. Indicates the battery is fully recharged.

B. Indicates the battery is supplying the DC bus.

C. Indicates the battery is partially discharged but still OPERABLE.

D. Indicates the battery is on the exponential charging current portion (the second part) of its recharge cycle

Proposed Answer: A. Indicates the battery is fully recharged.

Explanation:

A. Correct. According to the bases, "Required Action A.2 requires that the battery float current be verified as less than or equal to 2 amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, *it has now been fully recharged.* If at the expiration of the 12 hour period the battery float current is not less than or equal to 2 amps this indicates there may be additional battery problems and the battery must be declared inoperable in accordance with LCO 3.8.6 Required Action B.2."
B. Incorrect. If amps are greater than 2 amps, its an indication the battery is discharging.
C. Incorrect. Greater than 2 amps is an indications of discharging and the battery would be inoperable.

D. Incorrect. This is the bases for the minimum float voltage.

Technical References: B3.8.4 and B3.8.6

References to be provided to applicants during exam: None

Learning Objective: 9694H - Apply TS 3.8 Technical Specification bases

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

Difficulty: 3.4

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
013 A2.01 - Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; LOCA	**Group #**	1
	K/A #	013 A2.01
	Rating	4.8

Question 80

GIVEN:
- EOP E-3, Steam Generator Tube Rupture, is being performed
- SI and Phase A have been reset
- Charging is in service
- The crew has just completed step 26, ISOLATE Charging Injection

The Control Operator reports:
- RCS subcooling is 15°F and lowering
- Pressurizer level is 30% and lowering
- RCS pressure is 800 psig and lowering.

The Shift Foreman should direct the operator to __________1)__________ and go to _________________2)_______________________.

A. 1) reinitiate SI
   2) EOP E-1, Loss of Reactor or Secondary Coolant

B. 1) reinitiate SI
   2) EOP ECA-3.1, SGTR With Loss of Reactor Coolant, Subcooled Recovery Desired

C. 1) manually restart ECCS pumps as necessary
   2) EOP E-1, Loss of Reactor or Secondary Coolant

D. 1) manually restart ECCS pumps as necessary
   2) EOP ECA-3.1, SGTR With Loss of Reactor Coolant, Subcooled Recovery Desired

Proposed Answer: D. 1) manually restart ECCS pumps as necessary
2) EOP ECA-3.1, SGTR With Loss of Reactor Coolant, Subcooled Recovery Desired

Explanation:

A. Incorrect. Both are incorrect. E-1 is plausible as this is the procedure to deal with LOCA's. Actuate SI is the normal response to initiation of a LOCA/leak.
B. Incorrect. First part is incorrect. Second part is correct. ECA-3.1 deals with tube ruptures and another event, such as a LOCA.
C. Incorrect. First part is correct. Second part is incorrect.
D. Correct. Action per the Foldout page is to manually start ECCS pumps and go to ECA-3.1

Technical References: EOP E-3 step 26 and Foldout page.

References to be provided to applicants during exam: None

Learning Objective: 7336 - State contents of foldout page

Question Source: Bank #89 DCPP L121 08/2014 X

(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 08/2014	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 2.7

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	2
029 G2.1.32 - Containment Purge System: Ability to explain and apply system limits and precautions		**Group #**	2
		K/A #	029 G2.1.32
		Rating	4.0

Question 81

What is the reason for limiting the open time of the containment purge valves to no more than 200 hours for the year?

A. To prevent exceeding the NPDES permit.

B. To prevent valve erosion and subsequent excessive leakage past the valves when they are closed.

C. To minimize the probability of a LOCA occurring while the valves are open, limiting the offsite boundary doses.

D. To minimize the probability the valves will not fully close when the purge is secured, resulting in a breach of Containment integrity.

Proposed Answer: C. To minimize the probability of a LOCA occurring while the valves are open, limiting the offsite boundary doses.

Explanation:

A. Incorrect. The NPDES covers releases to the environment, but releases to the ocean, not from Containment.
B. Incorrect. These butterfly valves are not fully open (limited to 50 degrees of travel), so its plausible that there is concern of wear.
C. Correct .Containment purges rely on RM-44A and 44B to be OPERABLE. These are process rad monitors. Their operation is assumed to close the purge valves that are opened. Per ECG 23.3, The purging time restriction is meant to minimize the probability of a LOCA while conducting purging operations and thereby limit offsite boundary doses.
D. Incorrect. The valves are limited in travel because these valves have not been qualified to close under accident conditions. Knowing that there is a concern with closing makes this distractor plausible.

Technical References: CAP A-6A, ECG 23.3

References to be provided to applicants during exam: None

Learning Objective: 7428 – State gaseous radwaste system administrative controls

Question Source:	Bank #89 L162 01/2018	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 01/2018	Yes
Question History:	Last Two NRC Exams	Yes
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

Difficulty: 2.3

Examination Outline Cross-Reference	Level	SRO
	Tier #	2
035 A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the S/GS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure/level transmitter failure	Group #	2
	K/A #	035 A2.03
	Rating	3.6

Question 82

Unit 1 is at 100% power.

73 hours ago, Steam Generator 1-1 pressure channel PT-514 in Protection Set 1 - Rack 3, was inoperable and the actions to satisfy LCO 3.3.2 Condition D have been completed.

The Shift Foreman has just declared Steam Generator 1-4 pressure channel PT-544 in Protection Set 1 - Rack 3 inoperable due to a failed surveillance.

What action should be taken by the Shift Foreman?

A. Enter LCO 3.0.3

B. Be in MODE 3 in a maximum of 30 hours

C. Be in MODE 3 in a maximum of 78 hours

D. Trip the applicable bistables within 72 hours

Proposed Answer: D. Trip the applicable bistables within 72 hours

Explanation:

A. Incorrect. If there is a lack of familiarity with the bases statement about the note, then it could be thought that separate function(s) is for different functions, ie failed NI and failed level transmitter, both of which use ACTION D.
B. Incorrect. 30 hours is a typical time to MODE 3 in many ACTION such as C.2
C. Incorrect. If its assumed the bistables can't be tripped and so the shutdown action of D.2 applies.
D. Correct. Per the note, separate entry is allowed for each function. The Bases for LCO 3.3.2 states "In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. *When the Required Channels in Table 3.3.2-1 are specified (e.g., on a per steam line, per loop, per SG), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.*

Technical References: LCO 3.3.2 and B3.3.2

References to be provided to applicants during exam: first 3 pages of LCO 3.3.2

Learning Objective: 9697C - Apply TS 3.3 Technical Specification LCOs.

Question Source: Bank #
(note changes; attach parent) Modified Bank #

	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.2	

Difficulty: 2.7

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	2
002 G2.4.4 - RCS: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.		**Group #**	2
		K/A #	002 G2.4.4
		Rating	4.7

Question 83

GIVEN:

- Unit 1 is in MODE 4
- Pressurizer level is 20% and lowering
- RCS subcooling is 15°F
- The PZR PORVs and Safeties are all closed
- Train 'A' RHR is aligned for shutdown cooling

Which of the following procedures should be entered by the Shift Foreman?

A. EOP E-1, Loss of Reactor or Secondary Coolant

B. OP AP-24, Shutdown LOCA

C. OP AP SD-0, Loss of or Inadequate Decay Heat Removal

D. OP AP SD-2, Loss of RCS Inventory

Proposed Answer: B. OP AP-24, Shutdown LOCA

Explanation:
Shutdown AP's are not "major" procedures and therefore not RO knowledge

A. Incorrect. EOP E-1 is applicable for events at power
B. Correct. Applicable in MODE 4 for loss of RCS inventory
C. Incorrect. Applicable only in MODEs 5 or 6.
D. Incorrect. Applicable only in MODEs 5 or 6.

Technical References: LCO 3.3.3, B3.3.3, EOP E-1

References to be provided to applicants during exam: None

Learning Objective: 9694C - Apply TS 3.3 Technical Specification bases.

Question Source:	Bank #76 L111 11/12	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 11/12	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3.0

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	1
E04 EA2.2 - Ability to determine and interpret the following as they apply to the (LOCA Outside Containment): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	**Group #**	1
	K/A #	E04 EA2.2
	Rating	4.2

Question 84

The crew is performing the actions of EOP ECA-1.2, LOCA Outside Containment.

A few minutes after performing leak isolation steps, the following conditions exist:

- ECCS flow is stable
- Pressurizer level is 22% and rising slowly
- RCS pressure is 400 psig and lowering slowly

What procedure should be entered next by the Shift Foreman?

A. Go to EOP E-1.1, SI Termination.

B. Go to EOP ECA-1.1, Loss of Emergency Coolant Recirculation.

C. Go to EOP E-1, Loss of Reactor or Secondary Coolant, then go to E-1.1, SI Termination.

D. Go to EOP E-1, Loss of Reactor or Secondary Coolant, then go to E-1.2, Post-LOCA Cooldown Depressurization.

Proposed Answer: B. Go to EOP ECA-1.1, Loss of Emergency Coolant Recirculation

Explanation:

To answer this question, the SRO must know and understand the bases which explains what the expected response is if the LOCA is isolated and the appropriate procedure to be entered from ECA-1.2.

A. Incorrect. The bases for ECA-1.2 states: This step instructs the operator to check RCS pressure to determine if the break has been isolated by previous actions. If the break is isolated in Step 2, a significant RCS pressure increase will occur due to the SI flow filling up the RCS with break flow stopped. The operator transfers to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, if the break has been isolated, for further recovery actions. If the break has not been isolated, the operator is sent to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, for further recovery actions since there will be no inventory in the sump. However, going to E-1.1 is plausible because if the break is/was isolated, as they would do for a leak while in E-1.

B. Correct. Because pressure has not responded, the break is not isolated, ECA-1.1 is the appropriate procedure. The objective of the loss of ECR guideline is threefold: 1) to continue attempts to restore emergency coolant recirculation capability, 2) to delay depletion of the RWST by adding makeup fluid and reducing outflow, and 3) to depressurize the RCS to minimize break flow and cause SI accumulator injection.

C. Incorrect. The break is not isolated. The operator transfers to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, if the break has been isolated, for further recovery actions and then to E-1.1 if there are no other complications. If the break has not been isolated, the operator is sent to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, for further recovery actions since there will be no inventory in the sump.

D. Incorrect. The break is not isolated, therefore a transition to E-1 is not correct. However, if its thought the break is not isolated and the course of action is the same as a small RCS break, then the transition to E-1.2 would plausible.

Technical References: ECA-1.1 and 1.2 & background

References to be provided to applicants during exam: None

Learning Objective: 3552 - Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

Question Source:	Bank #80 L141 04/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 04/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3.5

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	1
APE 077 G2.4.30 Generator Voltage and Electric Grid Disturbances: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.		**Group #**	1
		K/A #	APE 077 G2.4.30
		Rating	4.1

Question 85

Unit 1 and Unit 2 are at 100% power.

Due to grid instability, the Shift Manager is in contact with the Grid Control Center (GCC) about an emergency backdown order.

Per Operations Policy B-1, Communications With Generation and Transmission Organizations, which of the following questions should be asked by the Shift Manager?

1. How quickly?

2. How many total megawatts need to be shed?

3. Adjustments to MVARs required?

A. 2 only

B. 3 only

C. 1 and 2

D. 1 and 3

Proposed Answer: C. 1 and 2

Explanation:

Ops Policy B-1 is implemented by the Shift Manager. The knowledge of the information that should be obtained by the SM when notified of the need to perform an emergency backdown.

A. Incorrect. 2 only The information that must be known is: 1. Is it an emergency, 2. How many total MW and 3. How quickly (time). The SM decides how much per unit and ramp rate (max is 100 MW/unit). 2 is one of the anwers, but how quickly (1) must also be asked.
B. Incorrect. 3 only. Plausible to think the amount of MVARS would be a concern for a backdown order and would be controlled by the GCC. Also, discussed in OP C-3:III, Main Unit Turbine – At Power Operations.
C. Correct. Of the questions listed, how quickly and the total amount needs to be asked. (Both 1 and 2)
D. Incorrect. 1 correct, 3 is not. Plausible to think the amount of MVARS would be a concern for a backdown order and would be controlled by the GCC. Also, discussed in OP C-3:III, Main Unit Turbine – At Power Operations.

Technical References: Ops Policy B-1, OP C-3:III

References to be provided to applicants during exam: None

Learning Objective: 3654 - Identify and discuss Operations Department policy statements

Question Source:

	Bank #81 L161 10/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 10/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.5	

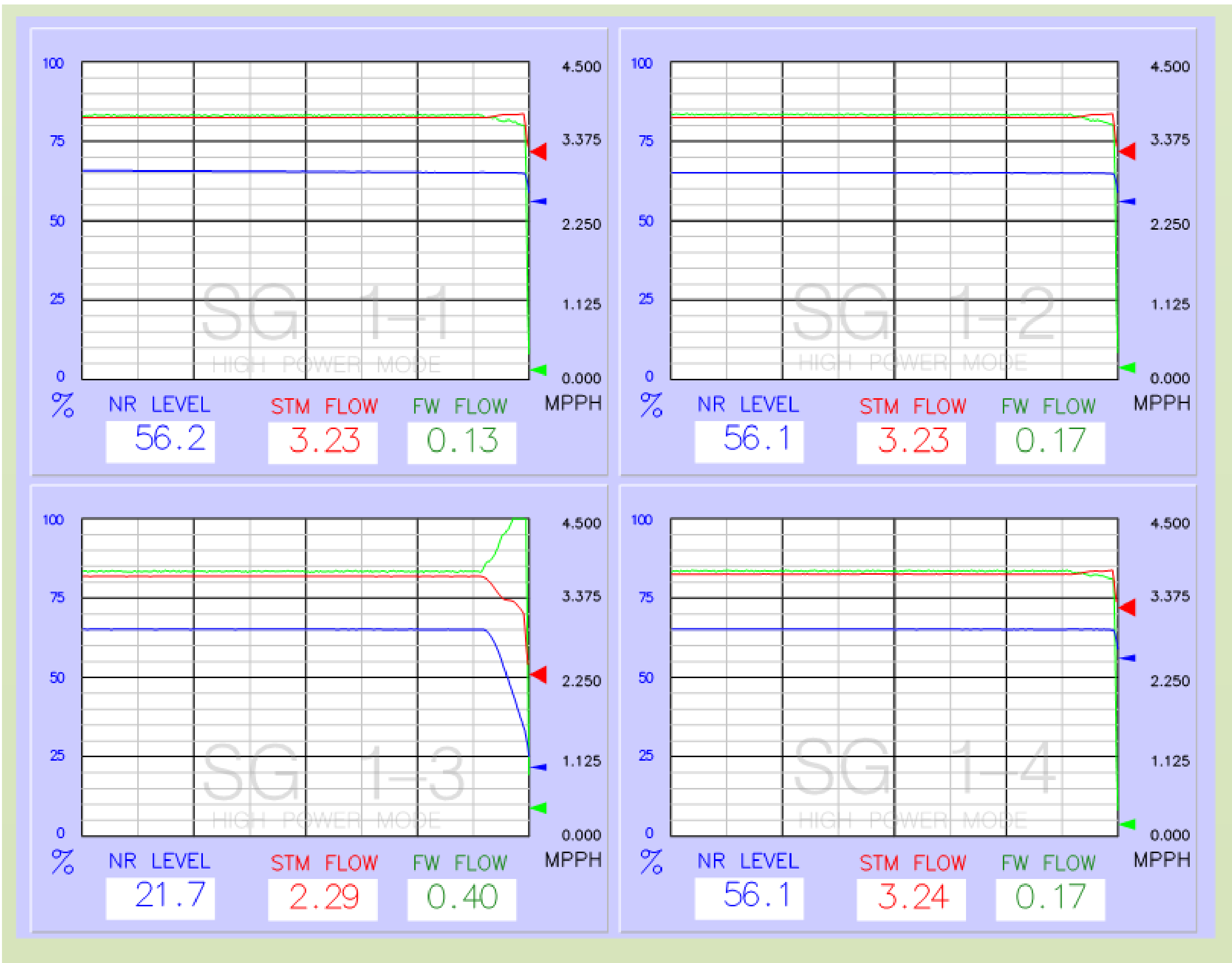
Difficulty: 2.2

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	1
APE 054 AA2.08 - Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Steam flow-feed trend recorder		**Group #**	1
		K/A #	APE 054 AA2.08
		Rating	3.3

Question 86

Unit 1 is at 100% power.

The following indications are seen on VB3:



1) What event is in progress?

2) In accordance with OP1.ID2, Time Critical/Sensitive Operator Action, what is the maximum amount of time the operator has to isolate the faulted steam generator?

A. 1) Main Steam line break
   2) 8.6 minutes

B. 1) Main Steam line break
   2) 10 minutes

C. 1) Main Feedwater line break
   2) 8.6 minutes

D. 1) Main Feedwater line break
   2) 10 minutes

Proposed Answer: D. 1) Main Feedwater line break 2) 10 minutes

Explanation:

A. Incorrect. Both parts incorrect. Steam flow would rise for a steam break. Time is to make a PORV available.
B. Incorrect. First part is incorrect. A feed break is in progress. Second part is correct.
C. Incorrect. Event in progress is a feed break. Time is plausible – 8.6 minutes is the time to make a PORV available for a feed break.
D. Correct. For a feed break, steam flow will lower as feed flow rises. For a steam break both will rise. Therefore, a feed break is in progress. In accordance with OP1.ID2, the time to isolate the faulted steam generator is 10 minutes.

Technical References: simulator. OP1.ID2
References to be provided to applicants during exam: None
Learning Objective: Explain the plant's response to a faulted S/G. (5461)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3.7

Examination Outline Cross-Reference	**Level** SRO
	Tier # 1
EPE 029 ATWS - G2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	**Group #** 1
	K/A # EPE 029 G2.4.9
	Rating 4.2

Question 87

Unit 1 is at 13% power.

The following sequence of events occur:

- A loss of 12 kV bus D occurs
- Operator attempts at a manual trip from the Control Room are unsuccessful
- The rods fully insert when the reactor trip breakers are tripped locally

1) An automatic reactor trip ___________________ have occurred when 12 kV bus D was de-energized.
2) In accordance with EP G-1, Emergency Classification and Emergency Plan Activation, is an Emergency Plan classification required for this event?

A. 1) should 2) No

B. 1) should 2) Yes

C. 1) should NOT 2) No

D. 1) should NOT 2) Yes

Proposed Answer: B. 1) should 2) Yes

Explanation:

A. Incorrect. Above 10% power, a loss of 2 RCPs should cause an automatic trip. Second part is incorrect classification is warranted.
B. Correct. Reactor trip is generated above 10%. Second part is correct. With actions in the control room to shutdown the reactor unsuccessful, an Alert is the proper classification.
C. Incorrect. First part is incorrect. Below 10%, loss of RCPs cause a reactor trip. Plausible because there are inhibits, such as C-5, which are active until 15% Second part incorrect, reactor was not shutdown by Control Room action.
D. Incorrect. First part incorrect, above 10%, a loss of 2 RCPs causes a reactor trip. Second part correct. Efforts to shutdown the reactor in the control room were unsuccessful.

Technical References: G-1 wall chart - EAL “HOT”, OIM B-6-3 and B-6-4a

References to be provided to applicants during exam: None

Learning Objective: 42285 - Given indications of an event, use EP G-1 to classify the event with 100% accuracy within 15 minutes

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.1	

Difficulty: 2.8

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	1
EPE 009 EA2.34 - Ability to determine or interpret the following as they apply to a small break LOCA: Conditions for throttling or stopping HPI	**Group #**	1
	K/A #	EPE 009 EA2.34
	Rating	4.2

Question 88

GIVEN:

- The crew has just entered EOP E-1, Loss of Reactor or Secondary Coolant from EOP E-0, Reactor Trip or Safety Injection
- Containment pressure peaked at 4 psig and is currently 2.3 psig and lowering slowly
- RCS pressure is 900 psig and rising slowly
- RCS subcooling is 45°F and rising slowly
- Narrow Range level in all steam generators are 18 to 22% and lowering slowly
- Total AFW flow is 440 gpm
- Pressurizer level is 25% and rising slowly

For the current plant conditions what:

1) procedure will the crew transition to from EOP E-1?

2) pump will be secured first?

A. 1) EOP E-1.1, SI Termination
   2) Charging pump

B. 1) EOP E-1.1, SI Termination
   2) Safety Injection pump

C. 1) EOP E-1.2, Post LOCA Cooldown and Depressurization
   2) Charging pump

D. 1) EOP E-1.2, Post LOCA Cooldown and Depressurization
   2) Safety Injection pump

Proposed Answer: A. 1) EOP E-1.1, SI Termination 2) Charging pump

Explanation:

SI termination criteria are:
RCS subcooling greater than 20°F
Secondary heat sink- either AFW flow above 435 gpm or narrow range level above 15% (25% adverse containment), RCS pressure stable or rising and pressurizer level greater 12% (40% adverse containment).

A. Correct. Both parts are correct. All criteria for terminating SI are met. The crew will transition to EOP E-1.1 and start by securing an ECCS charging pump.
B. Incorrect. First part is correct Second part is incorrect. SI termination criteria are met. A charging pump is secured first. Plausible to secure an SI pump first as it has a larger capacity than the charging pumps
C. Incorrect. First part is incorrect Second part is correct.

D. Incorrect. Both parts incorrect,

Technical References: EOP E-1, EOP E-1.1 EOP E-1.2

References to be provided to applicants during exam: none

Learning Objective: 3552 - Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #76 STP 2001	X
	New	
	Past NRC Exam STP #76 2001	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3.0

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	1
APE 026 G2.4.18 - Loss of Component Cooling Water:		**Group #**	1
Knowledge of the specific bases for EOPs.		**K/A #**	APE 026 G2.4.18
		Rating	4.0

Question 89

A design basis 200 gpm leak has occurred in the CCW system.

In accordance with OP1.ID2, Time Critical/Sensitive Operator Action, the leak must be detected and isolated within _____1)____ to prevent the loss of ____________2)___________.

A. 1) 20 minutes 2) CCW due to surge tank depletion

B. 1) 20 minutes 2) cooling to the charging pumps

C. 1) 1 hour 2) CCW due to surge tank depletion

D. 1) 1 hour 2) cooling to the charging pumps

Proposed Answer: A. 1) 20 minutes 2) CCW due to surge tank depletion

Explanation:

A. Correct. According to OP1.ID2, the time is 20 minutes to prevent surge tank depletion.
B. Incorrect. First part correct. Second part plausible as loss of CCW could result in a loss of cooling to CCW loads and the charging pumps have a TCOA associated with a loss of cooling (60 minutes to establish cooling from fire water).
C. Incorrect. First part incorrect. 60 minutes is the TCOA for establishing backup cooling to a charging pump. Second part is correct.
D. Incorrect. Both parts are incorrect.

Technical References: OP1.ID2 attachment 1

References to be provided to applicants during exam: None

Learning Objective: 3477L Given an abnormal condition, summarize the major actions of OP AP-11 to mitigate an event in progress including: Bases for TCOAs with operator actions of 1 hour or less.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.5	

Difficulty: 2.8

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	1
E10 EA2.1 Ability to determine and interpret the following as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.		**Group #**	2
		K/A #	E10 EA2.1
		Rating	3.9

Question 90

GIVEN:

- Unit 2 has tripped following a loss of condenser vacuum
- At 1300,
    - o the crew entered EOP E-0.2, Natural Circulation Cooldown
    - o CST level is 65%
- At 1700,
    - o CST level is 50%
    - o required boron concentration is established and the crew is ready to initiate an RCS cooldown from 547°F to MODE 4

Assuming the cooldown could be performed at the maximum rate allowed by EOP E-0.2, what action should be taken by the Shift Foreman?

A. Establish approximately a 25°F/hour cooldown and remain in EOP E-0.2.

B. Establish approximately a 50°F/hour cooldown and remain in EOP E-0.2.

C. The cooldown rate will have to be greater than the maximum allowed rate of 25°F/hour, transition to EOP E-0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS).

D. The cooldown rate will have to be greater than the maximum allowed rate of 50°F/hour, transition to EOP E-0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS).

Proposed Answer: D. The cooldown rate will have to be greater than the maximum allowed rate of 50°F/hour, transition to EOP E-0.3, Natural Circulation Cooldown With Steam Void In Vessel (With RVLIS).

Explanation:

A. Incorrect. Plausible - This is the Unit 1 rate. If the curves are not used or misread and wrong unit number applied, this would be correct.
B. Incorrect. Plausible - Using the wrong curve could make it appear there is adequate volume.
C. Incorrect. Plausible - If the unit 1 rate is used, and the proper curve, this would be correct.
D. Correct. At the unit 2 rate, it will take approximately 4 hours, there is not adequate CST level (requires greater than 50%).

Technical References: IF-2, IF-4 and IF-5, E-0.2 Unit 1 and E-0.2 Unit 2

References to be provided to applicants during exam: IF-2, IF-4 and IF-5

Learning Objective: 3552 - Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event

Question Source:	Bank #89 L161	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 10/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3.7

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	1
APE 036 G2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.		**Group #**	2
		K/A #	APE 036 G2.4.2
		Rating	4.6

Question 91

Unit 1 refueling is in progress.

The Refueling SRO observes bubbles coming to the surface of the refueling cavity and suspects the assembly being lifted is damaged.

In accordance with OP AP-21, Irradiated Fuel Damage, __________________ is an/are additional symptom(s)?

1. Containment Evacuation automatically alarms
2. Z-Z tape indicates the full down
3. Auto stop of upward hoist movement

A. 2 only

B. 3 only

C. 1 and 2

D. 1 and 3

Proposed Answer: B. 3 only

Explanation:

A. Inorrect. This is not an entry condition and is used to verify an assembly has been fully inserted into its core location.
B. Correct. This is an indication and is the only one listed.
C. Incorrect. The tape is not an indication. Containment evacuation, unlike FHB alarm does not actuate on high radiation. Plausible if its believed it does.
D. Incorrect. Containment evacuation alarm must be manually activated.

Technical References: OP AP-21

References to be provided to applicants during exam: None

Learning Objective: 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.7	

Difficulty: 3.1

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	1
E08 EA2.1 - Ability to determine and interpret the following as they apply to the (Pressurized Thermal Shock) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	**Group #**	2
	K/A #	E08 EA2.1
	Rating	4.2

Question 92

GIVEN:

- Safety Injection actuated due to a steam break on one steam generator 10 minutes ago
- All ESF equipment started
- All Steam Generator Narrow Range levels are offscale low
- The TDAFW pump has tripped
- Tcold on the affected loop is 200°F and lowering
- Intermediate Range Startup Rate on both channels is +0.1 DPM and stable

What procedure should the Shift Foreman transition to from EOP E-0, Reactor Trip or Safety Injection?

A. EOP E-1, Loss of Reactor or Secondary Coolant

B. EOP FR-H.1, Response to Loss of Secondary Heat Sink

C. EOP FR-P.1, Response to Imminent Pressurized Thermal Shock Condition

D. EOP FR-S.1, Response to Nuclear Power Generation/ATWS

Proposed Answer: C. EOP FR-P.1, Response to Imminent Pressurized Thermal Shock Condition

Explanation:

A. Incorrect. There are challenges to critical safety functions that must be addressed first.
B. Incorrect. Motor Driven AFW pumps provide adequate flow to prevent a challenge to heat sink.
C. Correct. Temperatures are to the left of limit A and are a Red path, which is currently the highest priority
D. Incorrect. Positive IR SUR is a MAGENTA condition. Although Subcriticality is a higher priority, by the rules of usage, the red path on RCS integrity must be addressed.

Technical References: F-0

References to be provided to applicants during exam: None

Learning Objective: 37107 - Apply the Rules of Usage in EOPs for the CSFSTs and FRGs, including:
the six status trees
the priority of use of the status trees
the priority of use of the color of each CSF
when to monitor and/or implement the CSFSTs and FRGs

Question Source:	Bank #99 L061C 02/2009	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 02/2009	Yes

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 3.0

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	1
APE 024 G2.4.6 Emergency Boration: Knowledge of EOP mitigation strategies.	**Group #**	2
	K/A #	APE 024 G2.4.6
	Rating	4.7

Question 93

GIVEN:

- The crew is performing EOP E-0.1, Reactor Trip Response
- A 10% steam dump valve is leaking by
- RCS temperature is 540°F and decreasing at a rate of approximately 2°F/minute
- RCS pressure is 2235 psig and lowering slowly
- Pressurizer level is 35% and stable
- AFW flow is throttled to approximately 440 gpm
- Steam Generator narrow range levels are 7% and stable
- MSIVs are closed

Which of the following actions should be taken by the Shift Foreman?

A. Continue in EOP E-0.1 and refer to OP AP-6, Emergency Boration.

B. Continue in EOP E-0.1 and implement OP AP-6, Emergency Boration

C. Direct the operator to initiate Safety Injection and go to EOP E-0, Reactor Trip or Safety Injection.

D. Direct the operator to initiate Safety Injection and go to EOP E-2, Faulted Steam Generator Isolation.

Proposed Answer: B. Continue in E-0.1 and implement OP AP-6, Emergency Boration.

Explanation:

A. Incorrect. The strategy is to initiate OP AP-6 but it is implemented, not referred to.
B. Correct. The proper action is to implement OP AP-6 and initiate emergency boration
C. Incorrect. Plausible to believe that SI is required and return to E-0 warranted. There are criteria for SI actuation in E-0.1, but none are met at this time.
D. Incorrect. Per the foldout page, if SI is initiated, the action is to go to E-0. There are procedures that direct going to EOP E-2 but in those procedures, SI has actuated and the diagnostic steps of E-0 have been performed. In EOP E-0.1, the SI steps must first be verified in EOP E-0. Additionally, the foldout page for SI actuation has not been met. There is adequate subcooling and pressurizer level.

Technical References: EOP E-0.1

References to be provided to applicants during exam: None

Learning Objective: 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event

Question Source:	Bank #99 L091 07/2011	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 07/2011	Yes

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.5	

Difficulty: 2.8

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	3
G2.1.34 Knowledge of primary and secondary plant chemistry limits.		**Group #**	1
		K/A #	G2.1.34
		Rating	3.5

Question 94

The crew has just completed a rapid load reduction of 20% in 30 minutes.

In accordance with LCO 3.4.16, RCS Specific Activity, a RCS sample must be taken within _________1)________ to determine if the limit for DOSE EQUIVALENT______2)______ has been exceeded.

A. 1) 2 to 6 hours — 2) I-131

B. 1) 2 to 6 hours — 2) Xe-133

C. 1) 8 to 10 hours — 2) I-131

D. 1) 8 to 10 hours — 2) Xe-133

Proposed Answer: A. 1) 2 to 6 hours 2) I-131

Explanation:

A. Correct. The potential for an iodine spike occurs when power is changed quickly (greater than 15% in one hour). The sample is drawn for Iodine within 2 to 6 hours, when the spike should occur.
B. Incorrect. First part is correct. Second part is incorrect, it is Iodine, not Xenon the sample is taken.
C. Incorrect. First part incorrect. The sample must be drawn within 2 to 6 hours. Second part is correct.
D. Incorrect. Both parts incorrect. Sample must be taken within 2 to 6 hours. Xe-133 is part of the LCO but not the reason the sample is required.

Technical References: LCO 3.4.16, SR 3.4.16.2

References to be provided to applicants during exam: None

Learning Objective: 9694D - Apply TS 3.4 Technical Specification bases

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

Difficulty: 3.0

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	3
G2.2.43 Knowledge of the process used to track inoperable alarms.		**Group #**	2
		K/A #	G2.2.43
		Rating	3.3

Question 95

In accordance with OP1.DC24, Control of Annunciator System Problems, the ______1)________ is responsible for conducting a ______2)_________ audit of annunciator problem logs.

A. 1) Shift Manager 2) weekly

B. 1) Shift Manager 2) monthly

C. 1) Shift Foreman 2) weekly

D. 1) Shift Foreman 2) monthly

Proposed Answer: A. 1) Shift Manager 2) weekly

Explanation:

A. Correct. Per the procedure, step 4.2, the SM is responsible conducting a weekly audit of the problem logs.
B. Incorrect. First part is correct. Second part is plausible as monthly is a normal amount of time for surveillances and reports
C. Incorrect. The SFM has many responsibilities and reviews, but the SM has the responsibility for reviewing the annunciator problem logs. Second part correct.
D. Incorrect. Both incorrect. The review is the responsibility of the SM and it is done weekly.

Technical References: OP1.DC24

References to be provided to applicants during exam: None

Learning Objective: 77187 - From memory, discuss the miscellaneous Ops department policies and administrative procedures per the guidance listed in the lesson references.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #69 L161 10/2017	X
	New	
	Past NRC Exam DCPP NRC 10/2017	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.3	

Difficulty: 2.1

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	3
G2.4.29 Knowledge of the emergency plan.	**Group #**	4
	K/A #	G2.4.29
	Rating	4.4

Question 96

In accordance with EP G-3, Emergency Notification of Off-Site Agencies,:

1) notifying the US Coast Guard of applicable Protective Action Recommendations is required at the ______ declaration.

2) the Coast Guard is notified after notifying the ____________________________.

A. 1) SAE
   2) county and state only.

B. 1) SAE
   2) county, state and NRC.

C. 1) GE
   2) county and state only.

D. 1) GE
   2) county, state and NRC

Proposed Answer: C. 1) GE 2) county and state only

Explanation:

PARs are declared for GE (unless dose assessment or field measurements exceed PAG criteria. This could be the case if the EOF is manned. If this was the case, the SM would not be making the call). Recent plant change to PARs removed any PAR below GE. Previously, at SAE PARs were possible.

A. Incorrect. USCG is not notified until after notifying the county and state, at the GE level. Desired outcome is all are notified within 15 minutes. This is done prior to notifying the NRC.
B. Incorrect. Required at the GE level, notification sequence is correct
C. Correct. USCG is notified after county and state at the GE level and prior to the NRC.
D. Incorrect. GE is the level of PAR declaration.County and state notifications made first and the Coast Guard is done prior to the NRC, not after.

Technical References: LEP-2, LEP-3, EP-G3, EAL wall chart (All page)

References to be provided to applicants during exam: None

Learning Objective: 42287 - As described in EP G-3, state the requirements for the following: Initial Notifications (Time Critical)

Question Source:	Bank #84 L141 04/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP NRC 04/2016	Yes

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.1	

Difficulty: 2.8

Examination Outline Cross-Reference		**Level**	SRO
		Tier #	3
G2.1.41 Knowledge of the refueling process.		**Group #**	1
		K/A #	G2.1.41
		Rating	3.7

Question 97

1) In accordance with ECG 42.1, Refueling Operations – Decay Time, prior to beginning a core offload, reactor must have been subcritical for a minimum of __________hours.

2) In accordance with OP L-6, Cold Shutdown/Refueling, unlatching RCCA drive shafts ___________ considered a CORE ALTERATION.

A. 1) 72 2) is

B. 1) 72 2) is NOT

C. 1) 100 2) is

D. 1) 100 2) is NOT

Proposed Answer: C. 1) 100 2) is

Explanation:

A. Incorrect. First part is incorrect. 72 hours is the time for many Technical Specifications to reach MODE 5, additionally a fuel handling accident is assumed to occur 72 hours after shutdown. Second part is correct.
B. Incorrect. Both parts are incorrect.
C. Correct. ECG 42.1 requires a decay time of at least 100 hours. OP L-6 states: NOTE: Unlatching RCCA drive shafts IS a CORE ALTERATION.
D. Incorrect. First part is correct. Second part incorrect.

Technical References: OP L-6, ECG 42.1

References to be provided to applicants during exam: None

Learning Objective: 66070 - Apply the requirements of System 42 ECGs.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
	Past NRC Exam	No
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.7	

Difficulty: 2.3

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	3
G2.2.37 Ability to determine operability and/or availability of safety related equipment.	**Group #**	2
	K/A #	G2.2.37
	Rating	4.6

Question 98

GIVEN:

- A degraded or non-conforming condition that might have an impact on applicable Systems, Structures, or Components (SSC) operability has just been reported to the Shift Foreman
- The SSC has a Technical Specification LCO ACTION completion time of 8 hours

The Shift Manager determines that a Prompt Operability Assessment (POA) is required.

In accordance with OM7.ID12, Operability Determination:

1) While the POA is being performed, the SSC is considered to be ________________.

2) The POA must be completed within __________________.

A. 1) inoperable 2) 8 hours

B. 1) inoperable 2) 24 hours

C. 1) OPERABLE 2) 8 hours

D. 1) OPERABLE 2) 24 hours

Proposed Answer: D. 1) OPERABLE 2) 24 hours

Explanation:

A. Incorrect. For a POA to be done, the immediate determination must be *OPERABLE* but more detailed investigation, evaluation or analysis is necessary to demonstrate the basis. The POA must be completed within a time frame based on its associated completion time. If less than 24 hours, the POA must be done within 24 hours. Plausible to think the CT is the time for the POA (as it is for CTs times between 24 and 72 hours).
B. Incorrect. For a POA to be performed, the determination is the expected condition is OPERABLE. Plausible because it may be viewed that the conservative action is to declare the equipement inoperable until the POA is complete.
C. Incorrect.The SSC is considered OPERABLE while the POA is performed. Plausible, the CT is correct for POA with times of greater than 24 hours.
D. Correct. The SSC is considered OPERABLE while the POA is performed. Additionally, the time for the POA is 24 hours if the POA is less than 24 hours.

Technical References: OM7.ID12

References to be provided to applicants during exam: None

Learning Objective: Explain the use of Prompt Operability Assessments. (35481)

Question Source:	Bank #97 L161 10/2016	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 10/2016	Yes

Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	X
10CFR Part 55 Content:	55.43.3	

Difficulty: 2.5

Examination Outline Cross-Reference	Level	SRO
	Tier #	3
G2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, *containment entry requirements*, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	Group #	3
	K/A #	G2.3.13
	Rating	3.8

Question 99

Unit 1 is in MODE 1. Plant and radiological conditions are stable.

Operators are preparing to make multiple containment entries per RCP D-230, Radiological Control for Containment Entry.

1) ______________________ is responsible for authorizing the Containment Entry.

2) ______________________ is responsible for conducting the Containment Pre-Entry Tailboard

A. 1) Shift Manager 2) Shift Manager

B. 1) Shift Manager 2) Radiation Protection

C. 1) Shift Foreman 2) Shift Manager

D. 1) Shift Foreman 2) Radiation Protection

Proposed Answer: D. 1) Shift Foreman 2) Radiation Protection

Explanation:

A. Incorrect. Shift manager has overall responsibility of the units, however, the SFM authorizes the entry. RP, not the SM does the brief.
B. Incorrect. First part incorrect. Second part correct.
C. Incorrect. First part is correct. Second part incorrect. Plausble, per OP AP-31, the SM conducts the brief for rapid containment entry. For normal entry, RP does the brief.
D. Correct. SFM authorizes and RP does the brief.

Technical References: RCP D-230, OP AP-31

References to be provided to applicants during exam: None

Learning Objective: 3101 - Demonstrate the ability to assume the responsibilities of the Shift Manager

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #98 L161 10/2016	X
	New	
	Past NRC Exam DCPP 10/2016	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.4	

Difficulty: 2.0

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	3
G2.4.40 Knowledge of SRO responsibilities in emergency plan implementation.	**Group #**	4
	K/A #	G2.4.40
	Rating	4.5

Question 100

GIVEN:

- The Shift Manager has just declared an Alert
- The Emergency Operations Facility and the Technical Support Center are manned, however, they have NOT yet been activated and no turnovers have occurred

For the purpose of Radiological Assessment Sampling, workers will be exposed to radiation levels in excess of the 10 CFR Part 20 exposure limits.

For the current plant conditions, the ____________ can authorize the dose on an EP RB-2, Attachment 9.7, Emergency Exposure Permit.

A. Radiation Protection Manager

B. Shift Manager

C. Site Emergency Coordinator

D. Emergency Director

Proposed Answer: B. Shift Manager

Explanation: needs to be updated

A. Incorrect. The Shift Manger has Command and Control of the emergency response for the given plant conditions. Plausible that the RP Manager would be responsible for approving exposures.
B. Correct. Only the Emergency Director or in this case, the Shift Manager (acting ED) has the has the unilateral authority and non-delegable responsibility for authorizing an individual emergency worker to exceed normal 10 CFR 20 exposure limits. The SM retains this role until the TSC or the EOF activated.
C. Incorrect. The TSC is not yet activated, the SEC is not yet taken control..
D. Incorrect. The ED would be the person responsible if in command and control.

Technical References: EP RB-2

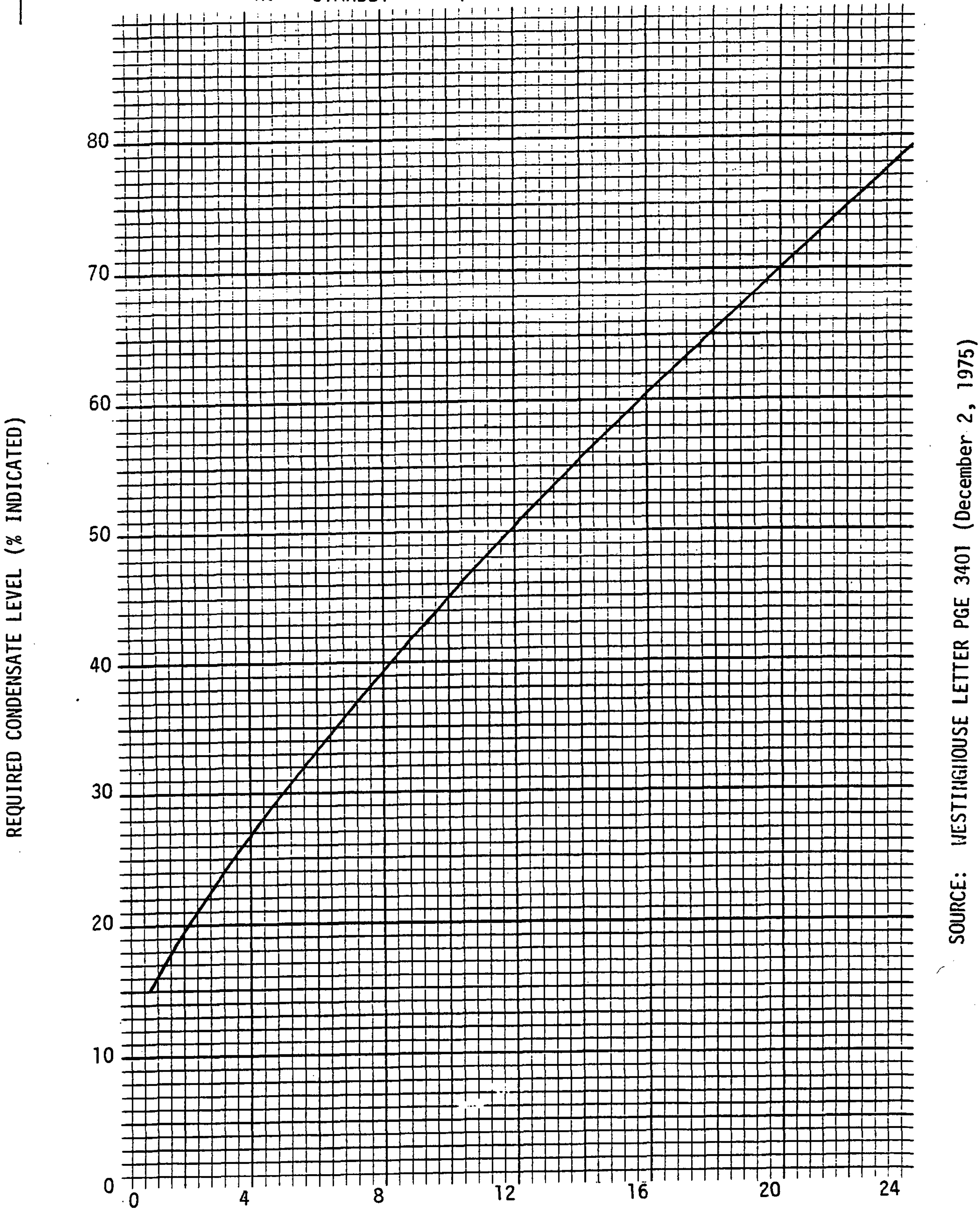
References to be provided to applicants during exam: None

Learning Objective: State who may authorize emergency doses. (9848)

Question Source:	Bank #97 L111 11/2012	X
(note changes; attach parent)	Modified Bank #	
	New	
	Past NRC Exam DCPP 11/2012	Yes
Question History:	Last Two NRC Exams	No
Question Cognitive Level:	Memory/Fundamental	X
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.1	

Difficulty: 2.3

DIABLO CANYON POWER PLANT OPERATION DATA
REQUIRED CONDENSATE LEVEL TO MAINTAIN A GIVEN
HOT STANDBY TIME (RATED MWt=3423)



HOT STANDBY TIME (HOURS) FIGURE IB-2

REVISION 1 DATE 4/6/81

DIABLO CANYON POWER PLANT OPERATION DATA
AUXILIARY FEED WATER REQUIREMENTS FOR VARIOUS
PERIODS OF HOT STANDBY AND PLANT COOLDOWN
FOLLOWING REACTOR TRIP SOURCE: DC 663200-56

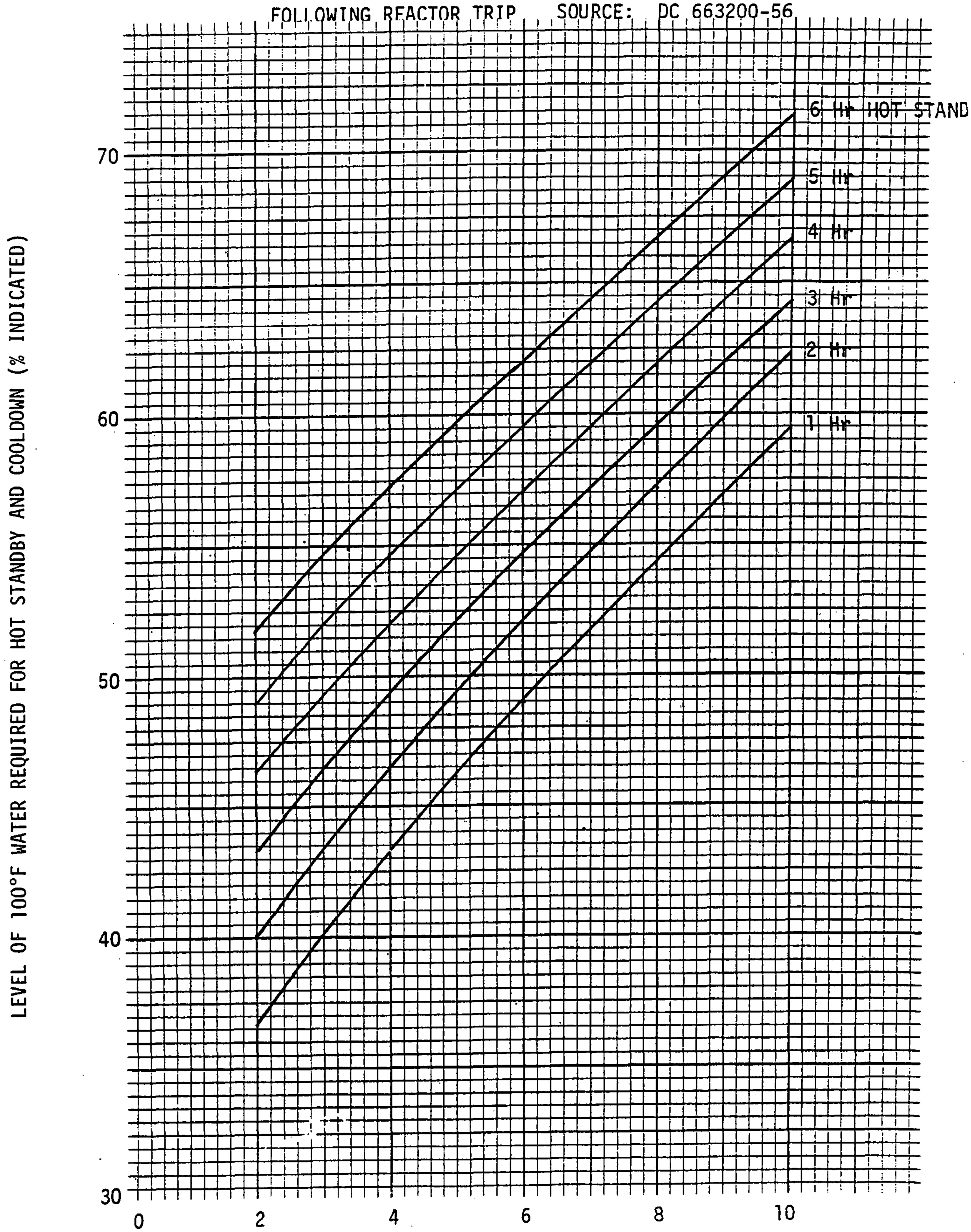


FIGURE IB-4

REVISION 0 DATE 6/3/77

DIABLO CANYON POWER PLANT OPERATION DATA
AUXILIARY FEEDWATER REQUIREMENTS FOR VARYING
PERIODS OF PLANT COOLDOWN STARTING A GIVEN
HOT STANDBY PERIOD (RATED MWt=3423) SOURCE: DC 663200-56

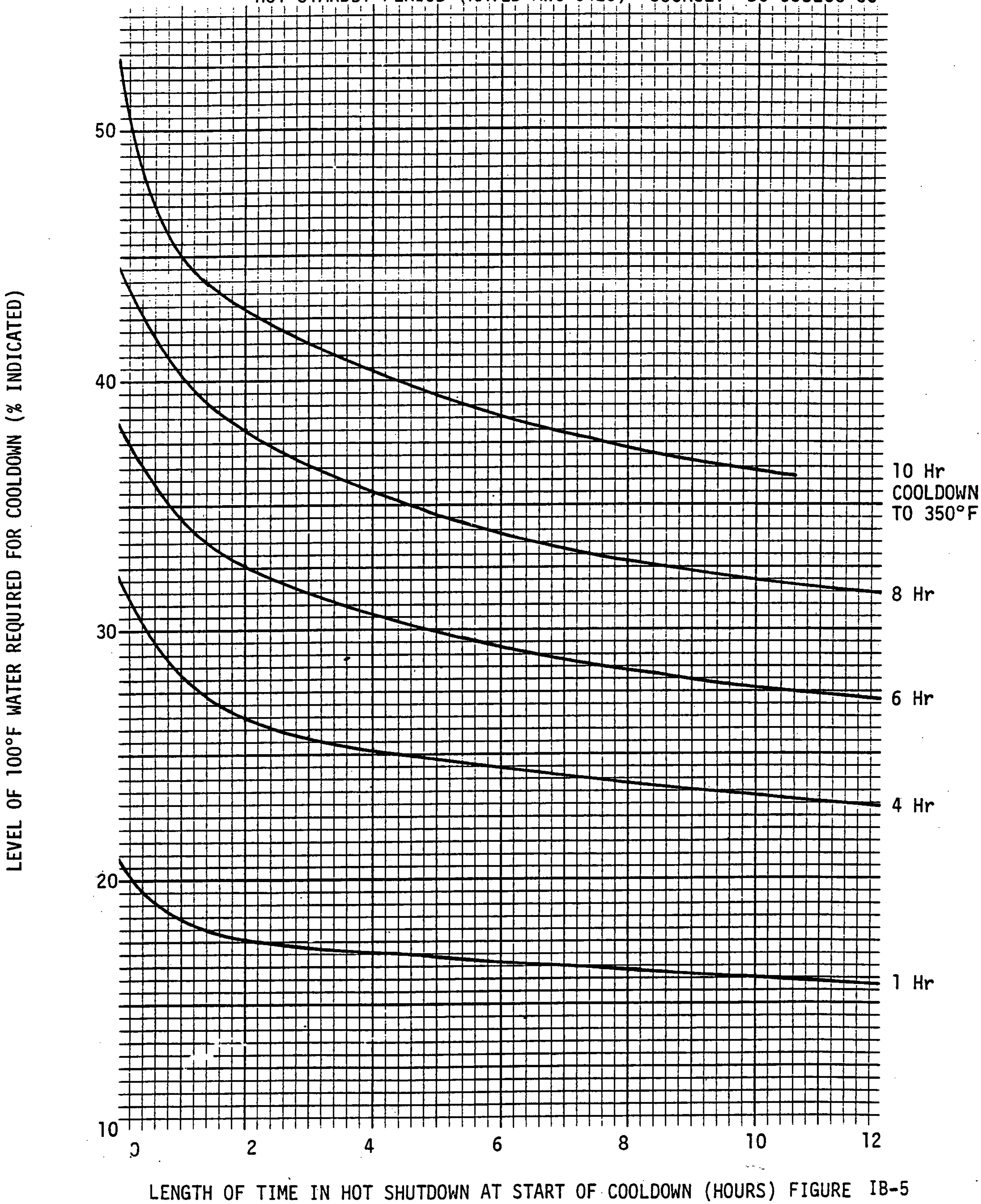


FIGURE IB-5

REVISION 0 DATE 6/3/77

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel(s) or train(s).	Immediately
B. One channel or train inoperable.	B.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2.1 Be in MODE 3.	54 hours
	<u>AND</u>	
	B.2.2 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------ Be in MODE 4.	60 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One train inoperable.	----------------NOTE----------------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---------------------------------------	
	C.1 Restore train to OPERABLE status.	24 hours
	OR	
	C.2.1 Be in MODE 3.	30 hours
	AND	
	C.2.2 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------ Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One channel inoperable.	------------------NOTE------------------ For function 1.d, the inoperable channel and/or one additional channel may be surveillance tested with one channel in bypass and one channel in trip for up to 12 hours, or both the inoperable and the additional channel may be surveillance tested in bypass for up to 12 hours. For functions 1.e(1), 4.d(1), 4.d(2), and 6.d(1), the inoperable channel and/or one additional channel may be surveillance tested with one channel in bypass and one channel in trip for up to 12 hours. This note is not intended to allow simultaneous testing of coincident channels on a routine basis. --	
	D.1 Place channel in trip.	72 hours
	<u>OR</u>	
	D.2.1 Be in MODE 3.	78 hours
	<u>AND</u>	
	D.2.2 Be in MODE 4.	84 hours

(continued)